NUCLEAR SCIENCE ABSTRACTS

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BIOLOGY AND MEDICINE

10803 UCLA-386
California. Univ., Los Angeles. Atomic Energy Project.
QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1956. Jan. 1, 1957. Decl. Mar. 14, 1957. 124p. Contract AT-04-1-GEN-12. $18.30(ph OTS); $6.00 (mf OTS).

Progress is reported in the following studies: the induction of leukemia in mice following injection of a cell-free extract; biochemical studies of leukemic cells; the isolation of ribonucleic acid from yeast; methods for the complete analysis of the fatty acids of tissues or organs; the synthesis and analysis of unsaturated fatty acids; the effects of total-body irradiation on the 5-hydroxytryptamine level in rats; the effect of irradiation on the blood serum Cu and Mg level in burros; the tissue distribution of Dextran in irradiated rabbits; the development of tracer methods for testing liver function, kidney function, thyroid function, and general metabolism; a comparison of physical, chemical, and biological methods for x radiation dosimetry; mechanisms of bone deposition of Y^{91}, Ca^{45}, and S^{35}, and Sr^{90} in rats and rabbits; effects of irradiation on porphyrin compounds; the production and isolation of an antibiotic agent; studies on the late effects of x irradiation on rats; routine health physics activities; the development of chemical radiation dosimeters for medical use; the development of a vacuum tube model of a time-of-arrival-of-fall-out-indicator; the metabolism of Ca and Sr in vertebrates; the redistribution of Pu in soil of a previously contaminated area; and an evaluation of sampling methods for the determination of α air concentration. (For preceding period see UCLA-379.) (C.H.)

AEROSOLS

10804 KAPL-M-JEL-1
[Knolls Atomic Power Lab., Schenectady, N. Y.]
A REVIEW OF BERYLLIUM AND URANIUM AIRBORNE CONTAMINATION IN KAPL SHOP AND LABORATORY AREAS. J. E. Loy. Jan. 17, 1951. Decl. Jan. 28, 1957. 9p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80 (mf OTS).

RADIATION HAZARDS AND PROTECTION

10805 CF-56-1-194
Oak Ridge National Lab., Tenn.
APPLIED HEALTH PHYSICS SEMI-ANNUAL REPORT FOR JULY 1955 THROUGH JANUARY 1956. J. C. Hart. 22p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Statistical data from area and personnel monitoring operations are summarized. (B.J.H.)

10806 HW-20628
Hanford Works, Richland, Wash.
RECENT CHANGES AFFECTING THE SPECIFICATIONS CONCERNING ACTIVITY LEVELS IN RECLAIMED URANIUM. C. C. Gamertsfelder. Mar. 27, 1951. Decl. Jan. 21, 1957. 2p. $1.80(ph OTS); $1.80(mf OTS).

CHEMISTRY

10807 KAPL-M-PHK-1
Knolls Atomic Power Lab., Schenectady, N. Y.
THE EFFECT OF TEMPERATURES UP TO 210°C ON THE RATE OF OXIDATIVE REMOVAL OF HYDROGEN FROM LINEAR POLYETHYLENES. Philipp H. Klein. May 2, 1956. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The rate at which hydrogen is removed by oxidative processes from a "linear" polyethylene has been studied at temperatures up to 210°C. The rate has been found to depend upon temperature according to the relationship Rate = 2.03 exp(-4050/T) in which the rate is given as the fraction of the hydrogen originally present in the polymer which is removed in an hour, and T in the temperature in degrees Kelvin. By analysis of activation-energy data, it is found that the diffusion of oxygen into the polymer is the rate-controlling step in the process. This fact leads to a simplification in methods of predicting oxidation-stability of bulk polymers at elevated temperatures. It should only be necessary to measure the actual rate of oxidation at one temperature and to determine the activation energy for the permeability of the material to oxygen in the desired range, in order to predict the over-all rate of oxidation. (auth)

10808 TID-5145
New Brunswick Lab., AEC, N. J.
EXPLORATORY STUDY OF THE SOLUBILITY AND DISTRIBUTION OF CERTAIN SALTS IN THE TBP–KEROSENE –URANIUM SYSTEM. Feb. 6, 1951. Decl. Mar. 1, 1957. 12p. $0.30(OTS).

10809 TID-7005
Bechtel Corp., San Francisco.
ENGINEERING RECORD BOOK—ZIRCONIUM CHLORIDE PLANT. Job No. 1788. Nov. 15, 1951. Decl. Feb. 13, 1957. 210p. For Northwest Electrodevelopment Lab. $53.30(ph OTS); $11.10(mf OTS).

10810 TID-10144
New Brunswick Lab. AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MARCH 31, 1954. C. J. Rodden. Decl. Mar. 13, 1957. 72p. $0.55(OTS).

Moving Bed Green Salt Process. Methods of preparing UO_3 feed for moving bed operations were examined. In the preparation of UO_3–H_2O slurries the settling time may be decreased by reducing the amount of H_2O, raising the surrounding temperature, or mixing the material at elevated temperatures (below 100°C). The addition of $UO_2(NO_3)_2 \cdot 6H_2O$ (UNH) can be used to delay the setting of the UO_3. The viscosity of the slurry varies from 15 to 100 poises or greater during the setting period. Pellet strengths were found to reach a maximum after about 8 hr of curing at room temperature. Feed materials containing 25 to 50 wt. % UNH could be pelleted, but they appeared either too soft on denitration or too weak after reduction to be used in the present moving bed reactor. The moving bed operations are described. Radiochemical Investigations. A rapid neutron absorption method for the determination of B^{10} in B metal has been developed. Analytical Chemistry. As the

results obtained in the Kjeldahl determination of N_2 in U derbies varied widely with the acid used for solution, work was initiated to determine which solution acids gave the best results. Solution with $HClO_4$ or H_2O_2 added to the initial HCl acid solution gave higher and more accurate results for N_2 in the derby metal than HCl alone. Colorimetric and spectrographic methods are presented for the determination of small amounts of SiO_2 in pure Mg. A method for the determination of rare earths as low as 0.01 ppm in a TBP process solution is summarized. Spectrochemical Investigations. A comparison of hand mixing vs. shaking as a means of adding Ga_2O_3 to U_3O_8 prior to spectrographic analysis showed that the precision of the hand mixed samples was higher. However, no conclusive results were obtained as to the relative merits of the two methods. (J.S.R.)

10811 TID-10153
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING SEPTEMBER 30, 1951. C. J. Rodden. Jan. 1952. Decl. Mar. 7, 1957. 101p. $0.65(OTS).

The distribution of U between 25% TBP–kerosene mixtures and 0 to 6M HNO_3 solutions was given. The U concentration studied varied from 0.2 g/l to saturation, and a nomograph was constructed for low U concentrations. The distribution of $Th(NO_3)_4$ between TBP–kerosene mixtures and 0 to 5M HNO_3 was studied. The thermal decomposition of $Th(NO_3)_4 \cdot 4H_2O$ was studied by heating the nitrate for varying lengths of time and analyzing the residue for ThO_2 and N_2O_5. The H_2 reduction of pelletized UO_3 was studied from 400 to 500°C using a thermobalance. The activation energy of the reaction $UF_6 + H_2 \rightarrow UF_4 + 2HF$, ~7100 cal, was determined from kinetic data obtained at 400, 476, and 538°C. The reactivity and density of lime and dolomitic limes ignited from 1500 to 3000°F were presented. The corrosion of U rods stored in outside cement vaults was shown to be the result of moisture condensation. Fourteen machined U billets were investigated to determine the depth of penetration of cracks, seams, and laps introduced during rolling. The hardness, grain size, dimensional change after β treatment, and preferred orientation were measured in an attempt to correlate the results of the different tests. The efficiency of the Rn counting chambers was calculated on a theoretical basis, and the calculated value was used to arrive at an estimation of the average percentage of the Po^{218} and Po^{214} deposited on the walls of the chamber. A simple flowmeter for measuring UF_6 flow rates was described. An improved counting chamber for use in counting neutrons in the neutron absorption analysis was developed. The distribution of Eu between aqueous $Th(NO_3)_4$–HNO_3 solutions and TBP in Dispersol decreases with increasing ionic strength of the nitrates in the aqueous phase. A research quantometer was tested and the sensitivity was comparable to the photographic technique and the precision exceeded that obtained densimetrically. (T.R.H.)

10812 TID-10154
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1951. C. J. Rodden. May 1952. Decl. Mar. 7, 1957. 112p. $0.70(OTS).

Progress is reported on the following investigations: utilization of a glass mixer-settler apparatus using air agitation for the TBP extraction of $Th(NO_3)_4$; solubility of TBP and TBP–kerosene mixtures in HNO_3 solutions of varying concentration; direct fluorination of $Th(NO_3)_4$ with anhydrous HF; the determination of water in ThF_4; precipitation of ThF_4 from TBP–$Th(NO_3)_4$–water systems with anhydrous and aqueous HF; precipitation of ThF_4 with NH_4F–HF from TBP–$Th(NO_3)_4$ systems; distribution of tracer amounts of Y and Eu between TBP solutions and aqueous $Th(NO_3)_4$–HNO_3 solutions; investigation of the reaction between H_2SO_4 and crude MgF_2 for the preparation of HF; salt-bath corrosion of U; protective coatings for U slugs; cause and effect of the hot spots developed during the rolling of U rods; the development of a differential (and integral) pulse height discriminator, a multiple-range rate meter and a linear amplifier for operation with an end-sensitive photomultiplier tube–NaI crystal combination; determination of small concentrations of In by radioactivation; determination of TBP in aqueous solutions of $Th(NO_3)_4$; determination of free HNO_3 in the presence of $Th(NO_3)_4$; concentration of impurities present in $Th(NO_3)_4$ for "shotgun analysis"; colorimetric determination of nitrates in UO_3; tentative extraction–polarographic method for the determination of trace quantities of U in Th; a rapid control method for the determination of U_3O_8 in a HNO_3 slurry of pitchblende ore; separation of rare earths from Th by hexamethylenetetramine; analytical procedures for the determination of sulfates, phosphates, and chlorides in mantle-grade $Th(NO_3)_4$; effect of Si in Q-11 ores on the spectrochemical determination of Si; preparation of a Mg standard for spectrographic use; and construction of a cathode-ray tube densitometer. (T.R.H.)

10813 TID-10157
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1952. C. J. Rodden. Apr. 1953. Decl. Mar. 9, 1957. 93p. $0.60(OTS).

Progress is reported on the following: TBP extraction of Th from Th sludges and mantle grade $Th(NO_3)_4$; studies on the acid extraction of U from recycled MgF_2 wastes; separation of Ir using anion exchange resin; U ingot quality studies; determination of U^{235} content of U metal bars by γ counting; determination of U^{235} by a radioactivation method; experiments with $CaCO_3$ as a carrier for Ra; an intensity control for spectrographic analysis; and flame photometric determination of Na, Li, and K in Be solutions. (T.R.H.)

10814 TID-10159
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING SEPTEMBER 30, 1953. C. J. Rodden. Mar. 16, 1954. Decl. Mar. 7, 1957. 48p. $0.45(OTS).

Modifications of existing laboratory processes for the extrusion, dehydration and denitration of hydrated UO_3 pellets are discussed. Present start-up procedures for both the reduction and hydrofluorination moving bed reactors are described. A joint operation is described wherein the consecutive reduction and hydrofluorination of dehydrated UO_3 pellets were accomplished in a single multistage reactor. UF_4 pellets were rehydrofluorinated in the moving bed using 150% of the theoretical anhydrous HF. The adiabatic compression of $(CH_3)_2O$–BF_3 complex from 1.0 to 1.25 atm pressure will cause a temperature rise of 7°C (from 128 to 135°). Work was done on a method of isotopic analysis of B in which the B was released in the form of BF_3 from the $(CH_3)_2O$–BF_3 product of an isotope separation column. With the use of a cathode-ray tube densitometer, it is possible to determine B in U at concentrations of approximately 0.02 ppm of B. (T.R.H.)

ANALYTICAL PROCEDURES

10815 CF-56-10-12
Oak Ridge National Lab., Tenn.
STATISTICAL QUALITY CONTROL QUARTERLY REPORT FOR THE ANALYTICAL CHEMISTRY DIVISION, JULY

THROUGH SEPTEMBER, 1956. M. T. Kelley. 79p. Contract [W-7405-eng-26]. $12.30(ph OTS); $4.50(mf OTS).

Control charts and summaries of the control data are presented. Brief discussions regarding the analysis of samples from the Pilot Plant as well as the analyses of a sample of UO_3, designated as NBL No. 18, are included. Information which is pertinent to the determination of U by the fluorometric, spectrophotometric, stannous chloride–thiocyanate, and the volumetric, zinc–ceric sulfate methods is also presented. (auth)

10816 K-721

Carbide and Carbon Chemicals Div. K-25 Plant, Oak Ridge, Tenn.

PRECISION OF THE PHOTOGRAPHIC SPECTROCHEMICAL ASSAY OF URANIUM 235 AT LOW CONCENTRATIONS. T. Lee, A. L. Hallowell, and Lewis H. Rogers. Mar. 20, 1951. Decl. Mar. 1, 1957. 21p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

A series of experiments was carried out to determine the precision of the spectrochemical procedure for the isotopic analysis of U, especially at low U^{235} concentrations. Standards were prepared from UF_6 which had been assayed on the mass spectrometer with a 95% confidence interval of 0.5% of the amount present. For a determination consisting of 9 separate measurements, the best precision obtained at normal concentration on the 95% confidence interval was $0.711 \pm 0.112\%$ U^{235} (16% of the amount present). (auth)

10817 KAPL-M-VFC-1

Knolls Atomic Power Lab., Schenectady, N. Y.

ANALYSIS OF SAMARIUM OXIDE IN 304 STAINLESS STEEL–SAMARIUM OXIDE CERMETS. Vivian F. Consalvo and Joseph Rynasiewicz. June 14, 1956. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Interfering metal ions such as Fe, Cr, and Ni are separated from Sm by the Hg cathode electrolysis from a $HClO_4$ solution. The Sm is then precipitated with $(NH_4)_2C_2O_4$. The resulting $Sm_2(C_2O_4)_3$ is ignited and weighed as Sm_2O_3. (auth)

RADIATION EFFECTS

10818 ANL-5004

Argonne National Lab., Lemont, Ill.

DECOMPOSITION AND RECOMBINATION IN IRRADIATED STATIC WATER SYSTEMS AT HIGH TEMPERATURES. J. R. Humphreys, E. K. Abers and Y. Solomon. Mar. 1953. Decl. Feb. 23, 1957. 56p. Contract W-31-109-Eng-38. $0.30(OTS).

Static water samples were exposed to pile radiation in the graphite reflector blanket of the Heavy Water Reactor at Chalk River, Ontario. These water samples were contained in an AISI type 347 stainless steel autoclave. The data indicate that between 300 and 500°F the hydrogen concentration was primarily a function of stainless steel corrosion and organic impurity oxidation reactions. The oxygen concentration, being repressed by the resulting hydrogen excess, approached trace concentrations. The decomposition product concentrations at 140°F were influenced by variations in the ionic impurity concentrations, while the higher temperature operations (300 to 500°F) were much less sensitive both to the presence of inorganic impurities and to variations in their concentrations. Small amounts of organic impurities in the water appear to be completely oxidized upon exposure to reactor radiation. In the experiments described, the quantities of CO_2 generated by organic oxidation were independent of temperature in the range of 300 to 500°F. With presence of a vapor space in the autoclave, the total autoclave gas content was lower than in the all-liquid-phase runs. The difference between the theoretical stoichiometric oxygen content (calculated from the measured quantities of hydrogen, carbon dioxide, and nitrogen) and the measured oxygen content was greater in the case of all-liquid-phase runs than in runs containing a vapor phase. A small excess of hydrogen in the two-phase experiments between 300 and 500°F greatly suppressed the amount of oxygen in the system. A small excess of oxygen, under the same conditions, correspondingly repressed the hydrogen concentration. Initial combination rates of hydrogen and oxygen added to two-phase, steady-state irradiated water systems between 327 and 500°F were essentially constant and independent of temperature. The initial combination rate at 295°F was lower than rates observed in the higher temperature range. A general evaluation of the experimental data indicates that the vapor phase combination reaction of hydrogen and oxygen is appreciable. In runs with addition of hydrogen and oxygen the major portion of the total combination reaction occurred in the vapor phase. (auth)

10819 CF-56-8-198

Oak Ridge National Lab., Tenn.

RADIATION DAMAGE TO FREON. M. D. Silverman, B. O. Heston, and P. S. Rudolph. Aug. 27, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A radiation experiment on Freon-11 was made to determine the decomposition products and their amounts under conditions approximating those anticipated for the HRT. A Van de Graaff accelerator was used for this decomposition study, as it provides a 100-fold more intense radiation source than the 1100 Curie Co^{60} source which was not available for prolonged continuous exposures and which was experimentally inadequate. (auth)

SEPARATION PROCEDURES

10820 CF-56-5-101

Oak Ridge National Lab., Tenn.

POWER REACTOR FUEL PROCESSING STATUS REPORT FOR MAY 1956. R. E. Blanco and W. K. Eister. May 15, 1956. Decl. Feb. 15, 1957. 10p. Contract [W-7405-eng-26]. $0.25(OTS).

Fused Salt–Fluoride Volatility Process. Dissolution rates of a Zr-clad U–Zr fuel element in $NaF \cdot ZrF_4$ were affected by the rate of flow of HF through the melt and the position of the metal relative to the HF inlet. The solubility of NiF_2 in $NaF \cdot ZrF_4$ appeared to be less than 1 wt % at 600°C. At 1 wt % and higher concentrations, NiF_2 was a stable colloid. The complex $UF_4 \cdot 3NaF$ was decomposed when heated at 400°C in the presence of excess UF_6. The U content of the residue was about 22%. Decomposition was much slower at 300°C or in the presence of excess F_2. The residual U was effectively pentavalent, as a result of the release of F_2 during the heating. Molten KHF_2 effectively decontaminated metals that had been exposed to molten fluoride mixtures containing fission products. Zircex Process. The heat of reaction in the hydrochlorination of Zr caused a rapid temperature rise. The temperature was controlled by introducing controlled amounts of H_2 with a constant HCl gas flow. The Zircaloy-2 cladding was chemically removed from a U–Mo alloy in laboratory studies. More than 99% of the U was obtained in a water solution separated from the insoluble molybdic acid remaining after dissolution of the core alloy. Hermex Process. Uranium that had been irradiated to an nvt of 5×10^{17} and decayed for 10 days was decontaminated from gross γ and β activi-

ties by factors of 20 and 70, respectively, by dissolution in Hg and distillation of the mercury. Continuous distillation of the Hg was shown to be possible. Sodium Amalgam Reduction of UCl_4. Uranium chloride was reduced to UHg_4 by sodium amalgam with a yield of approximately 80.5%. (auth)

10821 CF-56-8-176

Oak Ridge National Lab., Tenn.

THOREX PILOT PLANT, SUMMARY OF EQUIPMENT FAILURES DURING RUN HD-15. J. H. Walker. Aug. 20, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

10822 K-1013(Del.)

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

SEPARATION OF URANIUM HEXAFLUORIDE FROM BROMINE AND BROMINE TRIFLUORIDE. A. I. Krakoviak. June 21, 1954. Decl. with deletions Feb. 26, 1957. 15p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A procedure was developed for separating UF_6 from the reaction products of BrF_3 and U compounds. A product containing 99.99% UF_6 was obtained by solid condensation of the UF_6 from a 93% UF_6-7% Br mixture on the walls of a water cooled condenser at atmospheric pressure, followed by repeated flashing of this condensed product to another water cooled condenser; over 91% of the UF_6 in the original feed mixture was recovered. Similarly, a product containing 98.5% UF_6 was obtained from a 92% UF_6-8% BrF_3 mixture; over 80% of the UF_6 in the original feed mixture was recovered. (auth)

10823 K-1064

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

A BRIEF INVESTIGATION OF THE PARTITION OF PLUTONIUM AND URANIUM BY URANIUM PEROXIDE PRECIPITATION. G. F. Mills and H. B. Whetsel. Sept. 25, 1953. Decl. Feb. 23, 1957. 11p. Contract W-7405-eng-26. $0.25(OTS).

Under ordinary conditions of reaction, plutonium is carried on uranium when it is precipitated as the peroxide and no separation results. It has now been found that separation factors of 2 to 4 (based on plutonium-uranium ratios before and after precipitation) can be obtained if the plutonium is first oxidized to the plutonyl state and maintained there by suitable holding oxidants during the uranium peroxide precipitation operation. There is little doubt that better factors could be obtained by proper attention to variables revealed by this work. (auth)

10824 KAPL-M-JR-7

Knolls Atomic Power Lab., Schenectady, N. Y.

A LABORATORY PROCEDURE FOR THE RECOVERY OF EUROPIUM–OXIDE FROM EU_2O_3–STEEL CERMETS. Joseph Rynasiewicz. July 11, 1956. 7p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

10825 KAPL-M-MC-2

Knolls Atomic Power Lab., Schenectady, N. Y.

TOTAL PLUTONIUM ASSAY IN THE PRESENCE OF ALUMINUM BY $La(OH)_2$–LaF_3 CARRIER PRECIPITATIONS. M. Cefola and W. S. Andrus. Sept. 23, 1949. Decl. Feb. 23, 1957. 4p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

It has been established in many laboratories that the LaF_3 method for total Pu content in a solution containing Al ions gives incorrect results. This may be totally or in part caused by complex formation. Since many of the waste streams such as 1AW and 2AW in the Redox Process contain large amounts of $Al(NO_3)_3 \cdot 9H_2O$ it becomes quite important to have a method which will determine with some degree of accuracy the plutonium in these solutions. For unexplained reasons, substituting $La(OH)_3$ as a carrier precipitate, after having reduced the Pu to Pu(III) state by $NH_2OH \cdot HCl$, still gave low results even when the $La(OH)_3$ was converted to LaF_3 for mounting. The conversion was found necessary in order to eliminate large errors caused by the absorption of α particles by the former precipitate. The precipitation of $La(OH)_3$ by NaOH has the effect of separating Al from Pu by keeping the first in solution. A slight modification of the above scheme has been found to give satisfactory results in the presence of fairly large amounts of Al. Instead of using $NH_2OH \cdot HCl$ as a reducing agent, $SnCl_2$ in 6M HCl has been substituted. Precipitation of $La(OH)_3$ with NaOH should keep both Al and tin in solution by complex formation. (auth)

10826 M-3023

[Du Pont de Nemours (E. I.) & Co.] Jackson Lab., Wilmington, Del.

TENTATIVE DYE WORKS PROCESS FOR LEACHING C-100 (LOW SODIUM–HIGH CHLORIDE). C. W. Maynard, Jr. May 17, 1943. Decl. Mar. 7, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The chloride and boron content of C-100 ($Na_2O \cdot 2UO_3$) containing $\leq 3\%$ Na $\geq 0.02\%$ Cl is reduced by aqueous leaching, followed by filtration. C-100 containing up to 3% Na can be dissolved in HNO_3 and run through the extractor in the usual manner. If the Na content is much higher, it will crystallize out as $NaNO_3$. The chloride content of available C-100 may vary from 0.02% to 0.55%. The leaching operation reduces the chloride content to about 0.005%. (L.T.W.)

10827 M-5387

Linde Air Products Co., New York.

CAUSTIC CONSUMPTION IN PRECIPITATION OF URANIUM FROM TRICARBONATE SOLUTIONS. H. C. Given and R. F. Koontz. Jan. 20, 1947. Decl. Mar. 16, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

The amount of NaOH required in the precipitation of $Na_2U_2O_7$ from UO_2CO_3 was determined. (F.S.)

10828 MCW-222

Mallinckrodt Chemical Works, St. Louis.

EXPERIMENTS AIMED AT THE REMOVAL OF BORON FROM URANIUM COMPOUNDS. O. J. Buckheim. Aug. 1949. Decl. Feb. 28, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Attempts to volatilize appreciable quantities of B from uranyl nitrate liquors by esterification with various alcohols were unsuccessful. Boron was not removed from dry UO_3 by reaction with methyl alcohol. Polyhydric alcohols form nonvolatile complexes with B and apparently increase the distribution of B in ether. (J.E.D.)

10829 NAA-SR-1759

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JULY–SEPTEMBER 1956. E. E. Motta, A. G. Buyers, and W. J. Gardner. Jan. 1, 1957. Decl. Mar. 13, 1957. 32p. Contract AT-11-1-GEN-8. $0.35 (OTS).

Scale-up efforts for the high temperature processing of U fuels were concerned with relocated hot cell reconstruction and fittings. Heat transfer for induction coil cooling media has been correlated with power output for a 30 kw motor generator power supply. A fundamental, high temperature chemical study of the mechanism of fission product extraction by UO_2 in contact with molten irradiated U has been proposed. High temperature separations studies concerning the removal of fission products from irradiated

Th–U alloy have been directed toward the development of evaporative arc melting techniques, salt bed equilibration, and electrorefining from molten salt baths. Refractory slagging of irradiated Th–U alloy using, as dross, solid and liquid oxides and mixtures, represents a completed phase of the development study of the pyrochemical processing of Th fuels. A summary report of this investigation is included. Radiochemical analyses for fission products in solutions containing large amounts of Th and Ca have been improved. (auth)

10830 NYO-2048(Del.)
New Brunswick Lab., AEC, N. J.
SULFURIC ACID EXTRACTIVE DISTILLATION PROCESS FOR THE RECOVERY OF ANHYDROUS HYDROFLUORIC ACID FROM BY-PRODUCT AQUEOUS HYDROFLUORIC ACID. Preliminary Process Design and Economic Study. R. W. Legassie and J. P. Termini. Feb. 1954. Decl. with deletions Feb. 27, 1957. 29p. $4.80(ph OTS); $2.70 (mf OTS).

Laboratory and bench-scale developmental work at the New Brunswick Laboratory has established the technical feasibility of recovering anhydrous HF by the extractive distillation of aqueous 70 wt. % HF with H_2SO_4 in a packed tower. Anhydrous HF is removed as an overhead product while the bottoms, consisting of approximately 50 wt. % H_2SO_4 and less than 1 wt. % HF, is fed to a concentrator to produce a 50 wt. % H_2SO_4 for recycling. A preliminary process design and an estimate of the investment and operating costs are presented. The cost for concentrating aqueous 70 wt. % HF in this plant is estimated to range from $0.031 to $0.035 per pound of acid without amortization and from $0.047 to $0.053 per pound including a five year amortization. (auth)

10831 NYO-5114
[Mallinckrodt Chemical Works, St. Louis.]
WORK COMPLETED AT YALE ON AAA [PITCHBLENDE] ORE. Progress Report [Covering] the Period April 2 to May 11, 1945. Earl Miller, Howard Wibbels, and Harold Yeager. May 14, 1945. Decl. Mar. 8, 1957. 66p. $10.80 (ph OTS); $3.90(mf OTS).

A summary of the work done in establishing processing procedures for AAA [pitchblende] ores is described. Detailed experimental descriptions are given for each processing step. (W.L.H.)

10832 NYO-5115
Mallinckrodt Chemical Works, St. Louis.
ENGINEERING B DATA [PITCHBLENDE PROCESSING ENGINEERING]. Progress Report II [Covering the Period] April 2 to May 11, 1945. Earl L. Miller. May 15, 1945. Decl. Mar. 8, 1957. 18p. $4.80(ph OTS); $2.70(mf OTS).

Experiments were tried using a small Oliver filter and FEino test leaf. The results show that under the proper conditions both barium sulfate and gangue may be successfully handled on a continuous rotary drum type filter. The numerical rates obtained on the small Oliver were subject to considerable error because of cloth blinding. The FEino results should be used with allowance for the fact that the apparatus was available for only a few experiments. Washing experiments were tried but analytical results are not available at present. (auth)

10833 MonN-319
Clinton Labs., Oak Ridge, Tenn.
VARIABLES AFFECTING SOLVENT EXTRACTION AND A SUMMARY OF TWO CYCLE HEXONE EXTRACTIONS EMPLOYING IRRADIATED U-235. Problem Assignment Number TX5-11. Progress report [for] period Dec. 13, 1946 to May 1, 1947. F. R. Bruce. July 3, 1947. Decl. Mar. 2, 1957. 29p. Contract W-35-058-Eng-71. $7.80 (ph OTS); $3.30(mf OTS).

The effects of the following variables on the hexone extraction of uranium were investigated: temperature, and U, HNO_3, and salting-agent concentrations. Ruthenium was the main contaminant in the product resulting from two-cycle hexone extractions of Al–U^{235} alloys. (C.W.H.)

10834 NYO-5165
Mallinckrodt Chemical Works, St. Louis.
A STUDY OF MANNITOL IN HOT AQUEOUS URANYL NITRATE SOLUTIONS. T. C. Furnas. Feb. 15, 1946. Decl. Mar. 8, 1957. 7p. Contract [W-14-109-Eng-8]. $1.80(ph OTS); $1.80(mf OTS).

Experiments are presented on the use of mannitol to prevent the ether extraction of Mo from solutions of AAA (pitchblende) ores. (W.L.H.)

10835 NYO-5169
Mallinckrodt Chemical Works, St. Louis.
PURIFICATION OF V-4 SALT [URANIUM BEARING ORE] BY A NITRIC ACID DIGESTION AND ETHER EXTRACTION. James L. Miller. Feb. 25, 1946. Decl. Mar. 8, 1957. 12p. Contract W-14-109-eng-8. $3.30(ph OTS); $2.40(mf OTS).

Experiments were conducted to find a method of processing U bearing ores which could be applied to plant operation without changing the equipment used for processing other soda salts. It was proposed to use HNO_3 digestion adding $Fe(NO_3)_3$ during digestion as a salting agent. (W.L.H.)

10836 ORNL-2209
Oak Ridge National Lab., Tenn.
FEED MATERIALS PROCESSING STATUS REPORT FOR OCTOBER 1956. R. E. Blanco and W. K. Eister. Jan. 23, 1957. Decl. Mar. 14, 1957. 18p. Contract W-7405-eng-26. $0.30(OTS).

Metallex Process. Niobium was neither amalgamated nor dissolved by boiling in Hg at 356°C or by heating in fused $ZnCl_2$–KCl–NaCl matrix, containing $HgCl_2$, over Hg. In eight batch runs, made to produce Th quasi amalgam for metallurgical examination, a total of 2740 g of Th metal, as amalgam, was produced with an average yield of 84.5%. Fluorox Process. Oxidation at 600°C or higher of UF_4 produced by the Excer Process, which has a high specific surface area at 25°C, resulted in rapid sintering. Pelleting tests and two-stage reduction and hydrofluorination runs in moving-bed equipment with ore concentrate feed have been scheduled. In fluidization tests in which Excer UF_4 was oxidized, 60 to 150 mesh particles were the most satisfactory size. Excer Process. The density of $UF_4 \cdot {}^3/_4H_2O$ was increased from about 1 g/cc to about 3 g/cc by pressing and grinding. The uranium decontamination by electrolysis was as good when UF_4 was precipitated after electrolysis by adding HF to the UCl_4 electrolytic product as it was when the HF was present in the electrolytic cell and reduction and precipitation were simultaneous. The slurry-Hg separation equipment of the electrolytic cell was redesigned. Results of dehydration runs showed that about 80% water removal is possible with present equipment. Ore Processing. In preliminary tests high-pressure carbonate leaching resulted in extraction of more U and less silica than low-pressure. (auth)

SORPTION PHENOMENA

10837 CF-56-1-109
Oak Ridge National Lab., Tenn.
ADSORPTION OF H_2O BY ThO_2 AT HIGH TEMPERA-

TURES. D. M. Richardson. Jan. 20, 1956. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Adsorption studies were made to determine the existence of a stable hydrate of ThO_2. The classical volumetric method was chosen for the studies. Resulting adsorption data are shown graphically. (B.J.H.)

TRANSURANIC ELEMENTS AND COMPOUNDS

10838 CK-1512

[Chicago. Univ.] Metallurgical Lab.

CHEMICAL RESEARCH—SPECIAL CHEMISTRY OF PLUTONIUM REPORT FOR MONTH ENDING APRIL 1, 1944. Apr. 6, 1944. Decl. Feb. 16, 1957. 21p. (A-2093). $4.80(ph OTS); $2.70(mf OTS).

Data are presented on the solvent extraction of Pu compounds on a tracer scale. Samples of $Pu(C_5H_7O_2)_4$ have assayed 33 and 34% Pu. The theoretical value for $Pu(C_5H_7O_2)_4$ is 37.7% Pu. Treatment of $PuO_2 \cdot xH_2O$ with HBr at 700°C yielded a grass green $PuBr_3$. Various methods for producing Pu and properties of the product metal are described. (W.L.H.)

10839 CK-1756

Chicago. Univ. Metallurgical Lab.

SPECIAL CHEMISTRY OF 94—ANALYTICAL. Chemical Research Report for the Month Ending June 1, 1944. June 15, 1944. Decl. Feb. 16, 1957. 34p. (A-2463). $6.30(ph OTS); $3.00(mf OTS).

The copper spark technique for analysis of Pu is used on production samples and efforts to improve its sensitivity are reported. The use of cupferron for extraction of light metals from Pu is investigated using radioisotopes of Na, Be, and Sr. Other analytical methods for Pu studied are: $Th(NO_3)_4$ titration for F^-, Al morin fluorescence method for F^-, and spectrophotometric method for B. (T.R.H.)

10840 CN-3223(Del.)

Chicago. Univ. Metallurgical Lab.

VAPOR PRESSURE MEASUREMENTS ON PLUTONIUM METAL AND PLUTONIUM COMPOUNDS. T. E. Phipps, R. L. Seifert, and O. C. Simpson. Sept. 26, 1945. Decl. with deletions Feb. 15, 1957. 209p. Contract W-7401-Eng-37. $31.80(ph OTS); $9.30(mf OTS).

The apparatus and procedures used in vapor pressure measurements on Pu metal and Pu compounds are described. Properties of Ta which favored its use in the effusion vessel are discussed, and a schematic sketch of the effusion vessel is included. Data are tabulated on results obtained with Pu metal, PuF_3, $PuCl_3$, and $PuBr_3$. 27 references. (C.H.)

URANIUM AND URANIUM COMPOUNDS

10841 ANL-5255

Argonne National Lab., Lemont, Ill.

PURIFICATION OF NUCLEAR FUELS BY MELTING IN REFRACTORY OXIDE CRUCIBLES. Interim Report. H. M. Feder, N. R. Chellew, and M. Ader. Mar. 15, 1954. Decl. Feb. 7, 1957. 49p. Contract W-31-109-eng-38. $0.30(OTS).

Results are presented of a partially completed investigation of the phenomena occurring during the melting of irradiated uranium in refractory oxide crucibles. It is shown that under favorable circumstances certain fission product elements are separated to a considerable extent. Some of the separations achieved have been correlated by theoretical considerations. Application of this process to reactor technology is considered. Its use in connection with the enriched core of a fast neutron pile appears to be particularly attractive. (auth)

10842 ANL-RCV-SL-1120

Argonne National Lab., Lemont, Ill.

THE PROBLEM OF OXIDATION OF URANIUM IN ZPR-6. R. C. Vogel. Nov. 7, 1956. Decl. Mar. 9, 1957. 2p. Contract [W-31-109-eng-38]. $1.80(ph OTS); $1.80(mf OTS).

The effects of temperature on U oxidation rates in pure O_2 are shown. (D.E.B.)

10843 BMI-926

Battelle Memorial Inst., Columbus, Ohio.

ARC MELTING AND CONTINUOUS CASTING OF URANIUM. H. A. Saller, R. F. Dickerson, and E. L. Foster, Jr. July 13, 1954. Decl. Feb. 23, 1957. 32p. Contract W-7405-eng-92. $0.35(OTS).

Uranium has been successfully arc melted and continuously cast into the form of 3-in.-diameter ingots. Biscuit metal, production slugs, a 1 to 1 blend of machine chips and biscuit metal, and a 1 to 1 blend of machine chips and production slugs were melted into ingots which were evaluated by means of chemical analysis, metallographic analysis, and measurements of density and hardness. These studies indicated that machine chips do not produce high-purity metal. A metal procedure that would be satisfactory for use on a large production furnace was developed. This technique is based entirely on changing arc voltages, and therefore would be amenable to automation by means of electronic controls. This melting technique produced ingots with no shrinkage voids and with no unconsolidated areas. Molybdenum, graphite, and W electrode tips were studied. Molybdenum proved highly unsatisfactory for U melting. Tungsten tips contaminated the U to a level of about 300 ppm, and H_2O-cooled graphite tips were consumed at a rate of about 1.6 g/5 kg of metal melted. Metal produced by arc melting was compared with U produced by present production methods. (auth)

10844 FMPC-467

National Lead Co. of Ohio, Cincinnati.

REDUCTION OF UF_4 TO METAL CONTAINING UP TO 3% U-235. STANDARD OPERATING PROCEDURE FOR PILOT PLANT, METALLURGICAL AREA (SECTION 1.5.2.3). S. L. Reese. Sept. 1, 1954. Decl. Feb. 23, 1957. 13p. Contract AT(30-1)-1156. $0.25(OTS).

A detailed step-by-step description of the plant standard operating procedure is given. (B.J.H.)

10845 K-681

Carbide and Carbon Chemical Div. K-25 Plant, Oak Ridge, Tenn.

ELECTROCHEMICAL PREPARATION OF URANIUM TETRAFLUORIDE. PART II. HIGH TEMPERATURE CELL. R. W. Anderson, A. L. Allen, and E. W. Powell. Dec. 22, 1950. Decl. Feb. 26, 1957. 47p. Contract W-7405-eng-26. $0.45(OTS).

A continuous high-temperature process was developed for the preparation of UF_4 from UO_3. The process includes: preparation of a UO_2F_2–HF electrolyte by dissolving UO_3 in HF, precipitation of UF_4 hydrate upon reduction of UO_2^{2+} at a Hg cathode, thickening of the UF_4 slurry, filtration, washing of filter cake, and dehydration of UF_4 hydrate in a N_2 stream at elevated temperatures. (W.L.H.)

10846 K-831

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

FLUORINATION OF MASSIVE METALLIC URANIUM WITH LIQUID CHLORINE TRIFLUORIDE. T. S. McMillan, S. S. Kirslis, and E. J. Barber. Nov. 12, 1951. Decl. Feb. 26, 1957. 27p. Contract W-7405-eng-26. $0.30(OTS).

The influence of a number of variables upon the rate of reaction of liquid chlorine trifluoride with massive uranium metal has been investigated. At 30°C commercial liquid chlorine trifluoride reacts with metallic uranium yielding approximately 20 mg of uranium hexafluoride per square centimeter of geometric surface area of uranium per hour. Addition of hydrogen fluoride to the chlorine trifluoride was found to increase the rate of reaction by more than an order of magnitude. The concentration of hydrogen fluoride necessary for optimum reaction rate apparently depended upon the concentration of dissolved uranium hexafluoride in the fluorinating liquid. In the absence of uranium hexafluoride the reaction rate at 30°C was observed to be a maximum at a molar ratio of chlorine trifluoride to hydrogen fluoride of approximately 1.5 to 1; with the solution saturated with uranium hexafluoride, the maximum rate at 30°C was observed at a ratio of 0.28 to 1. Increasing the temperature of reaction was found to accelerate the rate of reaction, but the higher pressures necessary to keep the fluorinating mixture liquified made the fluorination more hazardous. Generally, addition of various metal fluorides to the fluorinating mixture had little effect on the rate of reaction. Addition of bromine trifluoride to a hydrogen fluoride–chlorine trifluoride mixture appeared to increase the rate to an appreciable extent, but technical difficulties inherent in handling such a mixture counterbalance any advantage gained from the higher rate of reaction. Rates of reaction corresponding to a penetration rate of about 0.007 centimeters of uranium per hour were obtained at 30°C with a fluorinating mixture saturated with uranium hexafluoride at 25°C. This rate will result in dissolution of a one inch diameter slug of uranium in about eight days and should be fast enough to furnish the basis for a volatility process for the separation of uranium from plutonium and fission products. (auth)

10847 K-847

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

CONVERSION OF URANIUM COMPOUNDS TO URANIUM HEXAFLUORIDE BY MEANS OF CHLORINE TRIFLUORIDE. PART III. FLUORINATION OF THE URANIUM OXIDES AT 60, 100, AND 140°C. R. L. Jarry and W. Davis, Jr. Dec. 21, 1951. Decl. Feb. 26, 1957. 21p. Contract W-7405-eng-26. $0.30(OTS).

Rates of conversion, in a dynamic system, of three uranium oxides (uranic, uranous, and urano-uranic) to uranium hexafluoride by means of chlorine trifluoride have been studied at 60, 100, and 140°C. At 60°C the initial rates for all three oxides were essentially the same, the rate for uranium trioxide falling off rapidly with time. At the higher temperatures this effect for the trioxide was considerably less. Analysis of the residues showed a composition tending toward that of uranyl fluoride in reactions involving uranous and urano-uranic oxide; residues from the uranium trioxide experiments were of the same composition as the original material. Products, other than uranium hexafluoride, were chlorine monofluoride, chlorine, chlorine dioxide, and chlorine oxyfluorides. Distillation analyses of these showed that as the oxygen to uranium ratio in the parent uranium oxide increases: 1) chlorine oxide formation increases; 2) efficiency of chlorine trifluoride utilization decreases. Stoichiometry calculations indicate that for the conversion of one mole of uranium to uranium hexafluoride from uranous oxide, urano-uranic oxide, and uranic oxide, 2.34, 3.03, and 4.84 moles of chlorine trifluoride are utilized. (auth)

10848 K-849

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

REACTIONS OF GASEOUS CHLORINE TRIFLUORIDE WITH URANIUM TETRAFLUORIDE. W. Davis, Jr. and R. L. Jarry. Dec. 18, 1953. Decl. Feb. 26, 1957. 38p. Contract W-7405-eng-26. $0.40(OTS).

The reaction of gaseous chlorine trifluoride with uranium tetrafluoride has been studied at 40, 60, and 80°C and at initial chlorine trifluoride pressures of 5 to 50 cm. Under these conditions the initial value of the mole ratio $(ClF_3)/(UF_4)$ varied from 2 to 20. To a first approximation the stoichiometry is given by equation 1 and the rate of conversion by equation 6, where t is the time, k a constant, z the number of moles of uranium hexafluoride formed from x_0 moles of uranium tetrafluoride: (1) $UF_4 + 0.9\ ClF_3 \rightarrow UF_6 + 0.7\ ClF + 0.1\ Cl_2$; and (6) $1-z/x_0 = e^{-kt}$. Within the pressure range studied the reaction is essentially independent of the chlorine trifluoride pressure. Average values of k at 40, 60, and 80°C are 0.551, 0.584, and 0.800 per hour, corresponding to 43%, 45%, and 55% conversion of unreacted solid per hour. The effect of temperature on the reaction is thus small. Uranium pentafluoride has been found in solid residues, indicating the reaction occurs, at least partially, by the sequence $UF_4 \rightarrow UF_5 \rightarrow UF_6$. (auth)

10849 K-1012

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

PREPARATION OF URANIUM HEXAFLUORIDE FROM URANIUM TETRAFLUORIDE USING BROMINE TRIFLUORIDE PREPARED IN SITU. E. B. Olszewski. May 7, 1953. Decl. Feb. 26, 1957. 18p. Contract W-7405-eng-26. $0.30(OTS).

A low temperature system for the continuous production of uranium hexafluoride from uranium tetrafluoride and/or the oxides of uranium using bromine trifluoride produced in situ as the fluorinating agent is described. Fluorine was added continuously to regenerate the reacted bromine trifluoride. Complete conversion to uranium hexafluoride was obtained at the maximum obtainable feed rate of 3.3 kg. of uranium tetrafluoride per hour. At feed rates of 0.44 to 2.14 kg. of uranium tetrafluoride per hour, fluorine efficiencies from 98.9 to 99.7 per cent were obtained. A method for the separation of uranium hexafluoride from the reaction products using crystallization in a cold water condenser is described. (auth)

10850 M-2948

[Du Pont de Nemours (E. I.) & Co.] Jackson Lab., Wilmington, Del.

TENTATIVE CHAMBERS WORKS PROCESS FOR THE MANUFACTURE OF CONCENTRATED URANIUM CONTAINING CAKES FROM DILUTE FEEDS "CONCENTRATION PROCESS." F. B. Stilmar. Oct. 16, 1945. Decl. Mar. 6, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

A process is described for the manufacture of concentrated U containing cakes from dilute feeds. (W.L.H.)

10851 M-3007

[Du Pont de Nemours (E. I) & Co.] Jackson Lab., Wilmington, Del.

TENTATIVE DYE WORKS PROCESS FOR THE MANUFACTURE OF C-106 FROM C-112 PASTE. Louis Spiegler. Apr. 20, 1943. Decl. Mar. 7, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

A tentative process for producing $UO_2(NO_3)_2 \cdot 6H_2O$ from $UO_4 \cdot 2H_2O$ paste is described. (T.R.H.)

10852 M-4563

Harshaw Chemical Co., Cleveland.

MOISTURE DETERMINATION IN GREEN SALT. K. S. Willson and W. R. Brennan. May 3, 1945. Decl. Mar. 7, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

These experiments show that moisture determination in UF_4 is somewhat elusive. However, it is concluded that some of the UF_4 supplied is wet and requires a prolonged drying period before going to the chemical reactors. (T.R.H.)

10853 MCW-218
Mallinckrodt Chemical Works, St. Louis.
DETERIORATION OF URANIUM TETRAFLUORIDE DURING STORAGE IN STEEL DRUMS. Carl W. Kuhlman. Sept. 14, 1949. Decl. Feb. 26, 1957. 3p. Contract [W-14-108-Eng-8]. $1.80(ph OTS); $1.80(mf OTS).

A lot of Plant 4 green salt was stored for $1\frac{1}{2}$ years and assayed periodically. Nonsignificant or only slightly significant change in assay was observed during this period. (D.E.B.)

10854 NYO-2040
New Brunswick Lab., AEC, N. J.
SUMMARY REPORT ON THE LONG RANGE PROGRAM FOR THE PERIOD FEBRUARY 16, 1953 TO APRIL 30, 1953. C. J. Rodden. June 24, 1953. Decl. Mar. 11, 1957. 78p. $0.55(OTS).

Additional kinetic studies have been made on the reduction of UO_3 and the hydrofluorination of UO_2 in the presence and absence of steam. Tail end reduction rates and hydrofluorination studies were made using a multiple tube reactor and a thermobalance. Further studies were made on the formation of pellets to be used in the moving bed reactor. A moving bed dehydrator was designed and the unit operated satisfactorily. Operation of the moving bed bench scale reactor was continued with certain modifications to the equipment. Studies on bomb reduction processes and the effect of slagging agents on the production of massive metal were continued. A thermodynamic study of slagging procedures is presented. Experimental work on the separation of powdered U from CaO using various leaching agents was continued. Results indicate that aqueous HNO_3 is the most suitable solvent. A pilot plant scale distillation tower was designed and constructed. A detailed description of the assembly and its component parts is included. (auth)

10855 NYO-5096
Mallinckrodt Chemical Works, St. Louis.
THE PROBLEM OF SPECTROGRAPHIC BORON DETERMINATIONS IN XO_2 [UO_2] AND X_3O_8 [U_3O_8]. B. D. Field. Mar. 29, 1945. Decl. Mar. 8, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

Discrepancies are discussed that were found in B determinations made by different laboratories. (L.D.)

10856 NYO-5126
Mallinckrodt Chemical Works, St. Louis.
A STUDY OF THE USE OF AMMONIA GAS TO REDUCE URANIUM TRIOXIDE TO URANIUM DIOXIDE. R. W. Shearer. June 12, 1945. Decl. Mar. 8, 1957. 16p. $3.30 (ph OTS); $2.40(mf OTS).

Results of experiments studying the use of NH_3 as a reducing agent in converting UO_3 to UO_2 indicate that cold, raw ammonia gas can be used in the reduction boxes in place of cracked ammonia gas and will reduce the trioxide to dioxide assay of 96.0% or better (gravimetric analysis). Indications are that it will take more raw NH_3 (roughly 20% more) than cracked NH_3 to obtain a given assay. There was no definitely distinguishable difference in reactivity between the UO_3 powder and milled cake. (J.E.D.)

10857 HW-28935
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
POSSIBLE PROCESS ALTERNATIVES FOR CONVERTING RECOVERED URANYL NITRATE TO URANIUM HEXAFLUORIDE. M. K. Harmon. June 23, 1953. Decl. Feb. 19, 1957. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

10858 TID-10160
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1953. C. J. Rodden. Mar. 19, 1954. Decl. Mar. 13, 1957. 43p. $0.40(OTS).

Development of the moving-bed hydrofluorination of UO_3 continued with production of a 180-lb. batch containing only 1.01 wt. % $(NH_4)_2C_2O_4$ insolubles and 98.9 wt. % UF_4. Reduction of UO_3 with H_2-N_2 and H_2-H_2O at 550 to 650°C was accomplished. A method of obtaining self-absorption corrections in Ra assay work is described. (T.R.H.)

WASTE DISPOSAL

10859 KAPL-M-WHT-1
Knolls Atomic Power Lab., Schenectady, N. Y.
ELIMINATION OF SECOND STAGE HOLD-UP OF LABORATORY LIQUID WASTE TO THE MOHAWK RIVER. W. H. Truran. Aug. 1956. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Certain KAPL liquid wastes, formerly subjected to evaporation, have for the period of June 1955 through June 1956, been released to the Mohawk River. The release procedure has involved two stage hold up and analysis followed by dilution with sanitary sewer effluent and storm water flows within the KAPL site. A review of operating procedures and experience suggests eliminating the second stage hold up and sampling. No loss of control over amounts of radioactivity or toxic chemicals in sewer effluent would result. Reduction in operating and maintenance expense would be significant. (auth)

ENGINEERING

10860 CF-56-10-16
Oak Ridge National Lab., Tenn.
HRT WASTE TANK EVAPORATOR SAMPLER. J. E. Kuster and C. A. Burchsted. Oct. 11, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A schematic of the aspirator-type sampler for HRT waste evaporators is given. Methods of preventing UO_2SO_4 freezing in suction lines are discussed. (D.E.B.)

10861 KAPL-M-S3G-RE-514
Knolls Atomic Power Lab., Schenectady, N. Y.
TRANSITION JOINT TEST; TEST NUMBER 514. J. Bianchi. Sept. 12, 1956. 25p. SAR Project. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

A shrink fit joint with a special type of thread was tested for use at the transition joint on the unit cell. The 514 A joint has sufficient strength for this application, being loaded to 34,000 lb without failure. (T.R.H.)

HEAT TRANSFER AND FLUID FLOW

10862 KAPL-M-EDL-59
Knolls Atomic Power Lab., Schenectady, N. Y.
NATURAL CONVECTION IN TOP HALF OF ROTATING PLUGS. W. T. Ahern and J. M. Marshall. Nov. 15, 1953. 31p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70 (mf OTS).

A study of temperature distributions in a shot-oil mixture was conducted in order to determine the effect of natural convection upon the effective conductivity of heat transfer in the shot-oil region above the cooling diaphragm of the

rotating plugs. A plot of temperature vs effective conductivity and temperature vs viscosity indicates that as the temperature rises, the effective conductivity is inversely proportional to the viscosity. For an approximate 50°F temperature drop with heat source temperatures of 150 and 225°F, Ke values of 117 and 179 Btu/hr-ft-°F respectively were obtained. Analysis of the temperature distribution with the oil lowered from contact with the cooling coils indicated that the oil temperature remained essentially constant throughout at steady state conditions. A considerable time lag in heating up the oil and shot mixture was observed at lowered oil levels. (auth)

10863 KAPL-M-RES-31
Knolls Atomic Power Lab., Schenectady, N. Y.
GENERAL TRANSIENT TEMPERATURE CURVES FOR ASTM A-302-B CARBON STEEL WALLS AND METHODS OF APPLICATION. H. F. Larkin. Sept. 5, 1956. 32p. SAR Project. Contract W-31-109-eng-52. $6.30(ph OTS); $3.00(mf OTS).

Curves are provided for transient temperature distributions in semi-infinite slabs and in ASTM A-302-B walls of various thickness for linear and non-linear temperature variations with time on the surface. In addition simple methods of solution are suggested to give a feeling of how complicated geometries may be handled in order to obtain answers within a useful range of engineering accuracy in a short time. (auth)

10864 KAPL-M-SMS-14
Knolls Atomic Power Lab., Schenectady, N. Y.
A TRANSIENT DYNAMIC ANALYSIS OF THE S3G/S4G NATURAL CIRCULATION STEAM GENERATOR SYSTEM. J. C. Westmoreland. Jan. 9, 1956. 28p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

A transient analysis has been performed on the proposed S3G/S4G natural circulation type steam generator, whereby the following dynamic characteristics of such a system undergoing rapid power output changes may be predicted: transient flow of liquid through separators, steam drum water-level surges, and distribution of liquid and vapor in system. The differential equations resulting from the analysis have been programed for numerical solution by the IBM-650 digital computer. Preliminary results on the three characteristics mentioned are presented and result from a prescribed pressure decay and steam flow increase, whose characteristics are also included in graphical form. The main consequences of rapid power maneuvers with these type units is noted to be the carry-over of moisture to the propulsion turbines which implies that adequate separator margin can not be provided for. The capability of the steam generator to supply adequate steam for any desired power maneuver has not been questioned, however it is noted that a reduction in downcomer flow during transient can reduce its capacity to do so. (auth)

10865 WAPD-PM-41
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
MIXING COEFFICIENTS. Henry Bohl, Jr., R. E. Grimble, and J. R. Parrette. Sept. 1955. 17p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

Mixing coefficients are used in the solution of heat transfer equations. (T.R.H.)

PUMPS

10866 CF-56-11-35
Oak Ridge National Lab., Tenn.
SUMMARY OF TESTS ON A PRODUCTION MODEL OF A BRYON JACKSON, 5-gpm PUMP IN LOOP DV 9. D. B. Weaver. Nov. 1, 1956. 11p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

WASTE DISPOSAL

10867 CF-56-10-15
Oak Ridge National Lab., Tenn.
WASTE AND DISPOSAL SYSTEM. J. E. Kuster. Oct. 1, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Waste and vent systems discussed in detail are venting and evacuation, flooding, liquid waste, evacuation of holdup tank, steam blowdown from heat exchangers, disposal, gaseous wastes, and waste storage emergency overflow. (M.H.R.)

10868 KAPL-M-DAM-1
Knolls Atomic Power Lab., Schenectady, N. Y.
RÉSUMÉ OF RADIOACTIVE WASTE HANDLING AT THE KNOLLS ATOMIC POWER LABORATORY. D. A. Manieri and C. N. Perleberg. May 1956. 6p. Contract W-13-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The history of waste handling operations at KAPL is reviewed. (M.H.R.)

MINERALOGY, METALLURGY, AND CERAMICS

CERAMICS AND REFRACTORIES

10869 CT-3373(Del.)
Massachusetts Inst. of Tech., Cambridge.
THE PREPARATION OF LARGE MAGNESIA CRUCIBLES. T. T. Magel, J. L. Ohlson, and A. Zais. Sept. 1945. Decl. with deletions Feb. 18, 1957. 26p. Contract W-7405-eng-175. $4.80(ph OTS); $2.70(mf OTS).

A method is described for producing MgO crucibles up to 4 in. in diameter and 10 in. high by machine tamping in steel molds moist (7% H_2O) fused magnesia powder. Green crucibles are air dried and then fired in an oxidizing atmosphere to 1700°C in catenary kilns lined with pure alumina bricks. The resulting crucibles are pure white, mechanically strong, given a bell-like ring when struck, and have a porosity of about 20%. Dimensions of the crucibles can be held to close tolerances (1 to 3%). (auth)

10870 CT-2433
Chicago. Univ. Metallurgical Project.
THE USE OF SULFIDE CRUCIBLES FOR REMELTING AND REDUCTIONS. L. Brewer, L. A. Bromley, N. Lofgren, P. Gillis, and W. Gwinn. Nov. 21, 1944. Decl. Feb. 19, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Th_4S_7, Th_2S_3, ThS, Ce_2S_3, Ce_3S_4, and CeS have been satisfactorily used for remelting of cerium metal in a vacuum at 1500°C. Results of tests are given and the effect of oxide impurities in the sulfide crucibles is considered. Cerium halide reductions have been carried out in CeS crucibles. If the CeS is not pure, the crucible is badly cracked. Pure CeS crucibles are satisfactory containers for the metal reduction reaction. (auth)

10871 WAPD-FE-434
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRELIMINARY LITERATURE SURVEY [ON] URANIUM–SILICON ALLOYS. K. M. Goldman. Aug. 20, 1954. Decl. Mar. 6, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

A survey of the mechanical properties, constitution diagrams, heat treatment, melting and fabrication characteristics, and powder metallurgical techniques of Si–U systems is presented. (J.E.D.)

CORROSION

10872 ANL-5380
Argonne National Lab., Lemont, Ill.
CREVICE CORROSION OF URANIUM AND URANIUM ALLOYS. J. W. Frank and A. H. Roebuck. Mar. 1955. Decl. Mar. 1, 1957. 39p. Contract W-31-109-eng-38. $0.40(OTS).

Three types of experiments were undertaken in order to study the corrosion of U and U alloys in high-temperature (500°C) water. Results of the capsule and bonded plate-pin hole experiments showed that corrosion of enclosed U is markedly different from that of U in open contact with H_2O. The gaseous hydriding experiments showed a definite relationship between rate of reaction and temperature, gas pressure, and alloy reacting. The corrosion of U in LiOH solutions was also studied. (C.W.H.)

10873 BM-II-92
Bureau of Mines. Northwest Electrodevelopment Lab., Albany, Oreg.
ADDITIONS ON THE HOT WATER AND STEAM CORROSION RESISTANCE OF ZIRCONIUM. G. L. Frederic, R. H. Robertson, and R. L. Carpenter. July 9, 1954. Decl. Feb. 12, 1957. 19p. $3.30(ph OTS); $2.40(mf OTS).

Neither niobium nor tantalum additions to zirconium or zirconium–1% tin alloys improved their corrosion resistance. Increasing the amount of niobium in these alloys resulted in increased rates of corrosion. Alloys of tantalum and zirconium had poor corrosion resistance. In the case of tantalum additions the presence of 1% tin tended to improve the resistance to corrosion, while the opposite effect was found in the niobium alloys. Niobium must be considered a deleterious element in a zirconium–1% tin alloy. None of the alloys tested proved equal to Zircaloy-2 in water at 600 or 680°F. The tantalum and tantalum–1% tin alloys were tested in steam at 750°F and a pressure of 1500 psi. The tantalum binary alloys all failed within 28 days, while the tantalum–1% tin alloys showed resistance comparable to Zircaloy-2 under these conditions. (auth)

10874 CF-56-2-40
Oak Ridge National Lab., Tenn.
SUMMARY OF LABORATORY CORROSION TESTS: HI-3 AND HI-4, CORROSION OF 63S ALUMINUM, IN CONTACT WITH LEAD, BY TAP WATER AND DEMINERALIZED WATER. E. L. Compere. Feb. 3, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Data sheets and photographs are given which describe the corrosion of 63S Al in H_2O. Conditions simulating possible situations in HRT instrument tubes were used. (B.J.H.)

10875 CF-56-2-100
Oak Ridge National Lab., Tenn.
HRP DYNAMIC CORROSION STUDIES: SUMMARY OF RUN J-76. 0.025 M SULFURIC ACID AT 150°C. J. C. Griess, R. S. Greeley, and S. R. Buxton. Feb. 20, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The effect of temperature on the corrosion of stainless steel by dilute sulfuric acid was extended to 150°C. Corrosion was much less at 150°C than at higher temperatures. Data from the run are summarized. (B.J.H.)

10876 CT-3052
Chicago. Univ. [Metallurgical Lab.]
CORROSION TESTING OF TUBALLOY [URANIUM]–COLUMBIUM ALLOYS. Frederick Nelson, W. W. Binger, and Joyce M. Hopkins. June 19, 1945. Decl. Feb. 18. 1957. 36p. Contract W-7401-eng-37. $6.30(ph OTS); $3.00(mf OTS).

Hot rolled and variously heat treated U-Nb alloys (2, 4, and 6 wt. %) were corrosion tested in relatively pure, air-free water. The corrosion resistance of similarly heat treated alloys in distilled water at 178°C increased with Nb content. Heat treatment also had an important effect on corrosion resistance. The effects of Nb content and of heat treatment on the corrosion resistance of the alloys in boiling, distilled water at 100°C were similar to those at 178°C. It is estimated that pimple growth on a leaky, unbonded, canned slug of the most resistent alloy, tested isothermally in water at 100°C, would be about 5 mils in 90 days. Air free, 1% NaCl solution attacked the water quenched 6 wt. % Nb alloy somewhat more rapidly at 178°C than did distilled water. This alloy was very severely attacked by air-free 1% NaF solution at the same temperature. Aerated water containing mixed electrolytes and a very small concentration of H_2O_2 attacked the water quenched alloys somewhat faster at 50°C than boiling water did at 100°C. (auth)

10877 KAPL-M-EGB-22
Knolls Atomic Power Lab., Schenectady, N. Y.
BEHAVIOR OF TYPE 347 STAINLESS STEEL IN SODIUM HYDROXIDE AT ELEVATED TEMPERATURES. E. G. Brush. July 12, 1956. Includes attachment: STRESS CORROSION CRACKING OF STEELS IN CAUSTIC SOLUTION. A LITERATURE SURVEY. E. L. Brady and E. G. Brush. Sept. 10, 1956. Addendum: by E. G. Brush. Sept. 14, 1956. 38p. Contract W-31-109-eng-52. $6.30(ph OTS); $3.00(mf OTS).

Tests have been performed in various NaOH mixtures to investigate expected behavior of type 347 stainless steel in the S2G third fluid NaK filled system in the event of a water leak. Idealized capsule tests show stress corrosion cracking to be severe in 100% NaOH and in NaOH solutions containing appreciable amounts of water. Tests of NaOH plus various additives show, however, that if the corrosion rate can be made to exceed 5000 mg/dm^2–mo the tendency to crack is reduced. The most promising additive seems to be NaH. (auth)

10878 KAPL-M-EGB-23
Knolls Atomic Power Lab., Schenectady, N. Y.
TRANSPORT OF RADIOACTIVE CORROSION PRODUCTS FROM CANNING DEFECTS. E. G. Brush. July 19, 1956. 18p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40 (mf OTS).

Theoretical calculations are made on the amount of transported activity arising as the result of corrosion through a small defect in a canned assembly placed in a neutron flux. They show that activity accumulates linearly after saturation until the canned material is consumed by corrosion, at which time standard decay mechanisms are effective. Analysis of behavior of a defected test assembly in a periodically interrupted neutron flux is also made, and provides a method of determining values for comparison with results of actual tests. (auth)

10879 KAPL-M-GEG-7
Knolls Atomic Power Lab., Schenectady, N. Y.
CORROSION OF CHROMALLIZED COATINGS IN 600°F WATER. G. E. Galonian, H. L. Tymchyn, and E. R. Harris. June 20 1956. 12p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The results of tests of chromallized coatings on carbon steel to determine corrosion resistance in 600°F water are described. Tests for 1, 4, and 12 weeks in deoxygenated, deionized water show that these coatings are corro-

sion resistant and protect the carbon steel base. The coatings investigated were applied by four different processes designated as XP 1030, XP 1039, XP 1040, and XP 1046. The first two showed the most promise for application in high temperature water. (auth)

10880 KAPL-M-UM-6
Knolls Atomic Power Lab., Schenectady, N. Y.
THE CORROSION BEHAVIOR OF SOME ZIRCONIUM–AND TITANIUM–BASE BORON ALLOYS IN HIGH TEMPERATURE WATER. Donald C. Belouin and Ulrich Merten. June 15, 1956. 12p. Contract W-31-109-eng-52. $3.30 (ph OTS); $2.40(mf OTS).

A number of Zr- and Ti-base alloys containing up to 2% B were exposed to high purity water at 600 and 680°F. The results indicate that a high degree of corrosion resistance is attainable in the Ti–B alloys and that the addition of 0.4% B to Zircaloy-2 increases its corrosion rate, but not drastically. (auth)

10881 WAPD-98
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
THE OXIDATION OF ZIRCONIUM AND ITS RELATIONSHIP TO CORROSION IN HIGH TEMPERATURE WATER. D. E. Thomas and J. Chirigos. Oct. 15, 1953. Decl. Mar. 9, 1957. 25p. Contract AT-11-1-GEN-14. $0.25(OTS).

The kinetics of oxidation of alpha zirconium in dry oxygen were determined and found to differ from the kinetics of corrosion in high-temperature water. The data suggest that the mechanism of reaction with the two media is similar except that an additional factor operates in water corrosion whose effect is not pronounced until the more or less protracted initial low corrosion rate is supplanted by a higher corrosion rate. (auth)

10882 MCW-161
Mallinckrodt Chemical Works, St. Louis.
CORROSION TESTS IN HF–H_2O ATMOSPHERE AT HIGH TEMPERATURE—NEGATIVE RESULTS. C. F. Ritchie. Jan. 12, 1949. Decl. Mar. 2, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

Procedure for testing the corrosion of 29 elements in HF–H_2O is outlined and results are tabulated. Preliminary tests showed that pure Al, Cu, and Mn were reasonably resistant to HF–H_2O mixtures at high temperatures. (J.E.D.)

METALS AND METALLURGY

10883 A-4049
Ames Lab., Ames, Iowa.
PRESENT PRACTICES IN THE URANIUM FOUNDRY AT IOWA STATE COLLEGE. C. F. Gray. Dec. 1945. Decl. Feb. 12, 1957. 35p. $6.30(ph OTS); $3.00(mf OTS).

10884 AGC-AE-22
Aerojet General Corp., Azusa, Calif.
THE REACTION OF MOLTEN METAL WITH WATER. Monthly Progress Report [for] July 1 through July 31, 1956. Sept. 14, 1956. Decl. Jan. 15, 1957. 9p. Contract AT(04-3)-44. $1.80(ph OTS); $1.80(mf OTS).

A discussion of previous and future work on molten metal drop tests, molten spray tests, high-pressure corrosion-rate tests, burning rates of metals in various atmospheres and transition-state theory of linear oxidation rates is presented. (W.L.H.)

10885 AGC-AE-26
Aerojet General Corp., Azusa, Calif.
THE REACTION OF MOLTEN METAL WITH WATER. Progress Report [for] August 1 through September 30, 1956. Oct. 26, 1956. Decl. Jan. 15, 1957. 10p. Contract AT(04-3)-44. $1.80(ph OTS); $1.80(mf OTS).

A study was made of the effects of the sudden mixing of molten metals, used in reactor fuel elements, with H_2O. Some data from explosion dynamometer tests are included. (B.J.H.)

10886 AGC-AE-28
Aerojet General Corp., Azusa, Calif.
THE REACTION OF MOLTEN METAL WITH WATER. Monthly Progress Report [for] October 1 through October 31, 1956. Nov. 23, 1956. Decl. Jan. 15, 1957. 12p. Contract AT(04-3)-44. $3.30(ph OTS); $2.40(mf OTS).

Tests have been performed to determine the reactivity of molten Al spray with degassed, deionized water at 725 to 2260°C. High-pressure corrosion-rate tests have been initiated on Zircaloy-2. The rate of propagation of burning of Mg in dry O_2 as a function of O_2 concentration has been determined. (For preceding period see AGC-AE-26.) (W.L.H.)

10887 ANL-5465(Del.)
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT [FOR] APRIL, MAY, AND JUNE 1955. Nov. 1955. Decl. with deletions Feb. 12, 1957. 72p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

The fabrication of Zircaloy-clad U alloy fuel elements, a Zircaloy-clad EBWR subassembly, Pu–Al fuel slugs for EBR-I, and resistance of U–Pu, to Na attack are reported. Work is reviewed on the melting and casting of U and Nb–U–Zr, properties of the UO_2–ThO_2 system, thermal conductivity, and corrosion of V and V alloys. The preparation of alpha U crystals, inclusions in U, C–U phase diagrams, phase studies of U–Zr and Nb–U–Zr and properties of Pu–U fissium alloys are presented. The mechanism of Zr corrosion, corrosion resistance of U alloys, and radiation effects on U compacts were studied. (J.E.D.)

10888 BM-II-72
Bureau of Mines. Northwest Electrodevelopment Lab., Albany, Oreg.
ZIRCONIUM PROGRESS REPORT FOR THE PERIOD OF JULY 15–AUGUST 15, 1953. Decl. Feb. 12, 1957. 23p. Contract AT(11-1)-140. $4.80(ph OTS); $2.70(mf OTS).

Production figures for Zr and Hf are reported. Data are presented on the evaluation of zircaloy ingots and corrosion testing of zircaloy and unalloyed Zr. Minerals from eight sources are evaluated for Zr, Hf, and rare metal content. (J.E.D.)

10889 CF-52-8-148 (Chap. 10)
Oak Ridge School of Reactor Technology, Tenn.
REACTOR MATERIALS. CHAPTER 10. G. E. Evans. Dec. 23, 1952. Decl. Feb. 16, 1957. 45p. $7.80(ph OTS); $3.30(mf OTS).

Stainless steels and their compositions and properties are described and tabulated. Corrosion behavior, impact strength, stress rupture, tensile strength, yield strength, creep, hardness, and elasticity are some of the properties included. (T.R.H.)

10890 CF-56-11-59
Oak Ridge National Lab., Tenn.
LINEAR THERMAL EXPANSION OF FOUR DIFFERENT COMPOSITIONS OF Be + BeO. J. E. Barton and S. D. Fulkerson. Nov. 7, 1956. 14p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Linear thermal expansion measurements were made on samples of hot-pressed Be containing 1, 1½, 2, and 3% BeO. Results of the measurements are presented in

graphical form and are tabulated. There was no uniform change in expansion with changes in oxide content between 1 and 3% in the samples tested. (B.J.H.)

10891 CT-685
Chicago. Univ. Metallurgical Lab.
TECHNOLOGICAL RESEARCH—METALLURGY: PART I OF REPORT FOR MONTH ENDING MAY 29, 1943. Decl. Feb. 18, 1957. 38p. (A-881). $7.80(ph OTS); $3.30(mf OTS).

10892 FMPC-488
National Lead Co. of Ohio, Cincinnati.
STANDARD OPERATING PROCEDURE FOR HEAT TREATING OF N.A.A. ENRICHED RODS. Section 1.4.3.13. Reed E. Yount. Nov. 11, 1954. Decl. Feb. 19, 1957. 8p. Contract AT(30-1)-1156. $1.80(ph OTS); $1.80(mf OTS).

The heat treatment and straightening of 2.75%, 0.85-in. diameter enriched U rods are described for the purpose of producing a beta structure and preparing the rods for machining. (J.E.D.)

10893 HW-29757
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
LABORATORY DEVELOPMENT OF AN ELECTROMAGNETIC TEST FOR URANIUM QUALITY. H. B. Headrick and E. C. Wood. Mar. 30, 1954. Decl. Feb. 25, 1957. 13p. Contract W-31-109-Eng-52. $0.25(OTS).

An electromagnetic instrument similar to the Multitest was built and used to test 2200 uranium slugs. It was found that the equipment could be used to detect inclusions and cracks in the uranium, but that small cracks are not readily detectable on a production basis unless the material is absolutely uniform in inclusion content. The instrument can be used, however, to sort out hidden defects of a reasonable size and to reject grossly dirty uranium. (auth)

10894 HW-31096
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
SOME CASTING VOIDS AND THE RESULTING DEFECTS IN ROLLED URANIUM. J. D. Hoffman, Jr. and R. L. Knecht. Mar. 1, 1954. Decl. Feb. 25, 1957. 15p. Contract W-31-109-Eng-52. $0.25(OTS).

Five U samples cut from the center of cast ingots and containing casting voids were reduced approximately 90% by rolling. The deformation of the voids during rolling was traced by radiographing and the resulting physical discontinuities in the rolled metal were microscopically examined. After a 90% reduction some of the casting voids were not completely closed; inclusions were concentrated inside or at the surface of many of the voids; and, apparently some of the void surfaces welded when pressed together while others did not. It is concluded that a casting void can result in a physical discontinuity running the entire length of an 8-in. slug. (auth)

10895 KAPL-1104
Knolls Atomic Power Lab., Schenectady, N. Y.
COLD-ROLLED AND RECRYSTALLIZED TEXTURES IN URANIUM SHEET; SOME EFFECTS OF TEMPERATURE AND RATE OF REDUCTION. W. Seymour and J. Duffey. Mar. 4, 1954. Decl. Mar. 29, 1957. 36p. Contract W-31-109-Eng-52. $0.35(OTS).

The preferred orientations of U sheet reduced 91% in a Sendzimir mill at room temperature were determined from x-ray intensity data. Complete pole figures for a large number of atom planes were plotted, and the preferred crystal orientation was found to be a mixture of (102) [010] and (012) [021], with the former predominating. The preferred orientation resulting from recrystallization was determined in a similar fashion and the ideal texture can be expressed as (100) [010]. It was found that sheet reduced 85% at 275°C had essentially the preferred orientation as the sheet rolled at room temperature, but sheet reduced 85% at 600°C had poles to (110) and (010) parallel to the rolling direction, with the poles to (110) predominant. A change in the rate of reduction by using rolling speeds of 100 and 200 ft/min at 275°C seemed to have little effect upon the final preferred orientation. However, some evidence was obtained which indicated that only a very small amount of material had poles to (010) parallel to the rolling direction when rolled at 100 ft/min at 600°C, but rolling at 200 ft/min at 600°C resulted in a slight increase of the (010) component of the texture. (auth)

10896 KAPL-M-AJC-1
Knolls Atomic Power Lab., Schenectady, N. Y.
MELTING POINT TEST CAPSULES. A. J. Ciancetta. Oct. 19, 1956. 10p. Contract [W-31-109-Eng-52]. $1.80 (ph OTS); $1.80(mf OTS).

The fabrication of the melting point test capsules is described. A program was established for finding methods of filling the capsules to obtain uniform distribution and a maximum density of UO_2. (B.J.H.)

10897 KAPL-M-JRP-1
Knolls Atomic Power Lab., Schenectady, N. Y.
COMPACTING AND SINTERING CHARACTERISTICS OF HORIZONS'S INC. AND CARBORUNDUM CO. ZIRCONIUM AND ELECTRO METALLURGICAL TITANIUM POWDERS. J. R. Pekoke. May 11, 1956. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Densities, hardness, and compression ratios for Zr and Ti powders are given for use in powder metallurgy processing. (T.R.H.)

10898 ORNL-551
[Oak Ridge National Lab., Tenn.]
BENDING OF A FUEL ASSEMBLY. W. B. Allred and H. C. Savage. June 9, 1948. Decl. Mar. 7, 1957. 12p. Contract [W-7405-eng-26]. (CF-48-6-174). $3.30(ph OTS); $2.40(mf OTS).

A study of the rigidity of a uranium fuel assembly to be used in the high flux reactor as proposed at Oak Ridge when considered as a beam is presented. The investigation includes the location and magnitude of bending and the resulting stresses that will be encountered when various uniformly distributed pressures are applied normally to the longitudinal axis. The mechanical stability of the assembly considered as a beam was investigated when uniform pressure loads were applied first, in a direction parallel to the surface of the active plates and second, in a direction normal to the surface of the active plates. In both cases maximum deflection and stresses occurred at the longitudinal center of the assembly. (auth)

10899 USBM-C-36
Bureau of Mines. Northwest Electrodevelopment Lab., Albany, Oreg.
ZIRCONIUM PROGRESS REPORT FOR THE PERIOD OF DECEMBER 15, 1954–MARCH 15, 1955. Apr. 4, 1955. Decl. Mar. 15, 1957. 37p. Contract AT(11-1)-140. $6.30 (ph OTS); $3.00(mf OTS).

The production plants produced 60,000 pounds of clean Zr sponge and 2,124 lb of Hf during the past quarter. A number of changes in the processing or reduction of Zr are described. Experiments show that H conditioning of Zr sponge before exposure to the atmosphere does not lower the O content in the finished product. In investigations of process improvement for Hf it was found that changing the chlorination procedure or substituting Ca for Mg in the re-

duction step had little effect on the O content. Further studies on the Ca deoxidation of Zr scrap show that the method is quite feasible but the recovery must be improved before the process can be considered economically satisfactory. A thorough study of consumable arc melting involving 1300 kg of material demonstrated that a recovery of 92.7% could be expected. The study of the mechanism of metallic arcs by means of high speed photography was continued and a number of results are presented. Hot water corrosion testing of Hanford process tubing indicates that the unalloyed Zr material does not have a long life expectancy in 600°F water. Other data regarding the effect of certain nuisance elements on the corrosion of Zircaloy-2 are shown. (auth)

10900 WAPD-107

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

RECOVERY OF IRRADIATED CRYSTAL BAR ZIRCONIUM. M. L. Bleiberg and L. S. Castleman. Apr. 23, 1954. Decl. Mar. 6, 1957. 27p. Contract AT-11-1-GEN-14. $0.35(OTS).

Annealed and 96% cold-worked crystal bar zirconium samples were irradiated for 3 to 6 months in a Hanford process tube at process water temperatures. The electrical resistivity (as measured at −196°C) increased a maximum of 21% for the annealed material and 25% for the cold-worked material. Isothermal annealing data for electrical resistivity were obtained for the irradiated specimens and compared to similar data for unirradiated cold-worked crystal bar zirconium. It was found that at lower temperatures the irradiated specimens recovered at a faster rate than the unirradiated cold-worked specimens. However, complete removal of the electrical resistivity increase due to irradiation required the same heat treatment which was needed to restore the original annealed resistivity value in 96% cold worked unirradiated zirconium. The electrical resistivity annealing behavior of the zirconium subjected to both cold work and irradiation could be predicted by assuming that the original increase in resistivity due to cold work still obeyed the annealing kinetics of cold-worked unirradiated zirconium, whereas the additional increase produced by irradiation obeyed the annealing kinetics of annealed irradiated zirconium. There were slight changes in hardness in both annealed and cold-worked zirconium due to neutron bombardment. The hardness annealing kinetics for 96% cold-worked zirconium appeared to be unaffected by neutron bombardment. (auth)

10901 WAPD-PWR-PMM-282

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

RÉSUMÉ OF URANIUM ALLOY DATA−X. E. F. Losco and Z. M. Shapiro. Nov. 14, 1955. Decl. Mar. 7, 1957. 67p. $0.50(OTS).

The principal developments in the PWR uranium fuel program consisted of the bond-line attack in coextruded Li−U system, Al-base alloys as barrier layers for U_3Si fuel elements, corrosion mechanism for bare Si−U systems, irradiation induced transformation in Mo−U and Nb−U alloys, and the UO_2 irradiation program. (J.E.D.)

10902 WAPD-PWR-FE-1192

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

DETERMINATION OF CAUSE AND MEANS OF PREVENTION OF WELD POROSITY IN PWR END CLOSURE WELDS. R. L. Fischer and A. H. Kasberg. Mar. 28, 1956. Decl. Feb. 7, 1957. 40p. $0.40(OTS).

Gross porosity, which frequently exceeded 60% of the cladding thickness, was found in the critical seal area of some of the early PWR development fusion welded end closures. The origin of porosity was traced to subsurface foreign material, principally located on the inside of drawn Zircaloy fuel tubes, which volatilized during welding. Removal of this foreign material by adequate pre-weld tube conditioning methods results in a reduction of weld porosity to a tolerable level. The amount of pre-weld conditioning required for fuel tubes has been found to vary considerably, and to have a strong correlation to the amount of conditioning employed during fabrication. (auth)

PHYSICS

10903 CF-56-10-136

Oak Ridge National Lab., Tenn.

ATMOSPHERIC PRESSURE SYSTEM FOR DETERMINING RESUSPENSION VELOCITY OF THORIUM OXIDE SLURRIES IN ROUND HORIZONTAL PIPES. D. G. Thomas. Oct. 10, 1956. 13p. Contract [W-7405-Eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

The literature on the subject was reviewed and discussed. A system for studying resuspension velocities was proposed and diagrammed. (M.H.R.)

GASES

10904 A-2352

[Standard Oil Co. of Indiana, Chicago.]

ANALYSIS OF VENT GASES FROM THE FRACTIONATION UNIT. D. A. McCaulay and W. L. Rittschof. July 24, 1945. Decl. Feb. 12, 1957. 20p. Contract W-7418-eng-41. $3.30(ph OTS); $2.40(mf OTS).

A method for the analysis of vent gases from the fractionation unit was developed which is satisfactory for control purposes. The method consists of separating the gas into two fractions, one condensable in liquid air and the other non-condensable. The condensables are analyzed by a novel procedure, while the non-condensables are analyzed on an Orsat apparatus. The accuracy of the method was established by analysis of synthetic mixtures containing all the components normally present in the vent gases. More than 1000 plant samples were analyzed on a routine basis with excellent results. (auth)

ISOTOPES

10905 CF-56-4-12

Oak Ridge National Lab., Tenn.

ESTIMATE OF THE COST OF PRODUCING NITROGEN 15 BY CHEMICAL EXCHANGE: AMMONIA−AMMONIUM CARBONATE SYSTEM USING CALCIUM HYDROXIDE REFLUX ASSUMING LOSSES OF 0.001 AT EACH REFLUX POINT. B. B. Klima and W. T. Ward. April 3, 1956. 45p. Contract [W-7405-eng-26]. $7.80(ph OTS); $3.30 (mf OTS).

The cost of producing one pound per day of N^{15} (95% isotopic purity), assuming a loss of 0.001 fluxer, is estimated to be $5,912. For a plant producing fifty pounds per day this cost drops to $1,827 per pound. These costs are approximately 47% and 69% higher respectively, than for plants of the same capacities designed on the basis of perfect (no loss) reflux. A breakdown of the investment costs, the fixed costs, and the variable costs are tabulated. A summary of annual operating costs including the cost per pound and per gram of N^{15} produced is given. (auth)

10906 CF-56-4-93

Oak Ridge National Lab., Tenn.

INVENTORY AND PRICE LIST OF ELECTROMAGNETI-

CALLY ENRICHED AND OTHER STABLE ISOTOPES. Apr. 1956. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

MASS SPECTROGRAPHY

10907 K-1049
Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.
A GAS ANALYSIS MASS SPECTROMETER. J. E. Skvarla. Jan. 5, 1954. Decl. Feb. 23, 1957. 27p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

A 60° General Electric isotope mass spectrometer has been modified for continuous mass scanning and recording. The spectrometer has been equipped with a magnet regulator which continuously changes the magnet current, a sample handling system permitting sample inlet through either a diffusive or a viscous flow leak, and a recorder. Procedures for using the instrument to identify components of many gas mixtures or to analyze them quantitatively have been developed. (auth)

MATHEMATICS

10908 KAPL-M-LFE-17
Knolls Atomic Power Lab., Schenectady, N. Y.
AN EQUATION OF STATE INVOLVING THE CRITICAL RATIO. Leo F. Epstein. May 4, 1956. 8p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

MEASURING INSTRUMENTS AND TECHNIQUES

10909 NAA-SR-Memo-488
North American Aviation, Inc., [Downey, Calif.].
THE USE OF THORIUM FOR FLUX MEASUREMENTS IN THE CONVERTER REACTOR MOCK-UP. D. H. Martin. Oct. 27, 1952. Decl. Feb. 27, 1957. 8p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

The possible use of Th for neutron flux measurements in the converter mock-up experiment was investigated. Foils prepared from Ames Th which is older than a few hours from preparation time are found to possess too much beta activity for low flux measurements. This beta activity is due to daughter products in the decay chain and may be reduced almost to background level by chemical purification. Activity curves for purified ThO_2 samples were obtained experimentally and a period of about 6 hours of constant low activity was observed starting at about 3 hours from purification. Foils have been made of purified ThO_2 and are found to have satisfactory characteristics on irradiation in the water boiler neutron source. (auth)

NEUTRONS

10910 LRL-84
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
SPECTRUM OF INELASTICALLY SCATTERED NEUTRONS. F. E. Bjorklund. Mar. 1954. Decl. Feb. 27, 1957. 10p. Contract AT(11-1)-74. $0.25(OTS).

A semi-empirically correlation of available inelastic cross section data in the energy region of interest for fast reactors has been developed. This correlation, which is based on nuclear evaporation theory, predicts an average spectrum of inelastically scattered neutrons for neutrons in the required energy range. Specific results are presented for several elements of interest in the fast reactor program. (auth)

10911 KAPL-M-GJH-3
Knolls Atomic Power Lab., Schenectady, N. Y.
THREE-SPACE-DIMENSIONAL MULTIGROUP. George J. Habetler. [nd]. 25p. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Finite difference equations for the numerical solution of the age diffusion equation in three space-dimensions are derived. The geometry is hexagonal in a plane and cartesian in the direction perpendicular to the plane. (auth)

10912 KAPL-M-NCF-1
Knolls Atomic Power Lab., Schenectady, N. Y.
AN IMPROVED DIFFUSION THEORY BOUNDARY CONDITION. N. C. Francis, J. C. Stewart, and P. F. Zweifel. Oct. 9, 1956. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

In diffusion theory calculations of neutron distributions in two adjoining media, the usual boundary conditions applied are the continuity of neutron flux and current. A new boundary condition is formulated to replace the flux condition. This condition is based on a rigorous transport theory solution for the net current of neutrons from one medium to the next, and is based on neutron conservation arguments. Application is made to the plane interface between two semi-infinite media as well as to the boundaries of a slab imbedded in an infinite medium. It is felt that the new boundary condition gives better answers for flux-peaking and self-shielding calculations than may be obtained from ordinary diffusion theory boundary conditions. In particular, a relation is derived between peaking and "effective self-shielding" which allows one of those factors to be calculated if the other can be measured or estimated. (auth)

10913 KAPL-M-RTF-5
Knolls Atomic Power Lab., Schenectady, N. Y.
CORRECTION OF MEASURED FLUX SCANS FOR FINITE RESOLUTION EFFECTS. R. T. Frost. Oct. 5, 1956. 14p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40 (mf OTS).

The influence of the finite spatial resolution of neutron flux scanning devices is discussed. Methods for correcting the measured distributions for this effect are given, and the results are particularized to the ATR double-bed scanner. It is shown that the corrections are, in general, quite small except when scanning near the end of an activated strip sample. The corrections should, however, be applied when the distributions are to be used to derive parameters such as thermal utilization, where high precision is required. (auth)

10914 KAPL-M-RWD-13
Knolls Atomic Power Lab., Schenectady, N. Y.
CALCULATION OF THE NEUTRON AGE IN HYDROGENOUS MIXTURES. R. W. Deutsch and I. A. Podhorez. [1955?]. 12p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The neutron age for a U^{235} fission spectrum has been calculated for water-metal and water-hydrocarbon mixtures. The materials calculated are Al, Fe, stainless steel, Zr, C, polystyrene, polyethylene and furfuryl alcohol. The procedure followed has been to assume the experimental water age and consider the relative change of the average transport and slowing down cross sections when other materials are added. There is good agreement with present available experimental information for metal-water mixtures. (auth)

10915 NYO-3923
Kidde (Walter) Nuclear Labs., Inc., Garden City, N. J.
THE ψ_3 APPROXIMATION TO THE TRANSPORT EQUATION FOR CYLINDRICAL GEOMETRY. Nunzio Tralli

and Karl Puechl. May 15, 1953. Decl. Mar. 28, 1957. 19p. Contract AT(30-1)-1374. (WKNL-10). $0.25(OTS).

The spherical harmonic method has been extended to cover anisotropic scattering and has been applied in various approximations to the problem of a cylindrical fuel rod surrounded by a water moderator. Both finite and infinite moderators are considered. An approximate procedure for the calculation of thermal utilization is also described. Calculations have been performed on lattices containing slightly enriched U fuel and H_2O moderator. The validity of the approximations used has been considered only for lattices of this nature. (auth)

NUCLEAR PROPERTIES

10916 KAPL-M-EFC-2

Knolls Atomic Power Lab., Schenectady, N. Y.

INELASTIC SCATTERING PARAMETERS OF IRON. E. F. Clancy. Sept. 27, 1956. 10p. Contract W-13-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Inelastic cross sections for iron as well as energy distributions of scattered neutrons have been determined, the latter quantities averaged over certain energy groups used in age calculations. Angular distributions have been assumed to be isotropic. (auth)

10917 KAPL-M-RWD-15

Knolls Atomic Power Lab. Schenectady, N. Y.

THE NEUTRON ABSORPTION CROSS SECTION OF Xe^{135}. R. W. Deutsch. July 20, 1956. 11p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

The absorption cross section has been recently deduced by Petruska, et al., based on a fission yield experiment in the NRX neutron spectrum. They calculated a value of 2.9×10^6 b. On the other hand, the total cross section measurement of Bernstein, et al., predict two possible absorption cross sections. These cross sections averaged over the assumed Maxwellian spectrum for the NRX yield values of $2.19 \pm 0.07 \times 10^6$ b and $2.45 \pm 0.07 \times 10^6$ b, respectively. In order to explain the disagreement, the data of the Petruska, et al., experiment has been reanalyzed. The value which would apply to the assumed neutron spectra of Petruska, et al., is found to be $2.78 \pm 0.13 \times 10^6$ b, which is in reasonable agreement with the set of parameters yielding the higher absorption cross section. The composite effect of unknown factors such as spectrum hardening, epithermal captures, and fluctuation of the flux during sample irradiations is believed to be small. It is, therefore, concluded that the set of parameters which best describe the absorption of Xe^{135} is $g = 5/4$, $E_0 = 0.0849 \pm 0.0010$, $\Gamma_n^0 = 0.0182 \pm 0.0050$, and $\Gamma_\gamma = 0.0942 \pm 0.0032$. (auth)

10918 WAPD-93

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

VARIATION THROUGHOUT REACTOR LIFETIME OF THE REGENERATION FACTOR, η, FOR SLIGHTLY ENRICHED URANIUM. W. R Clancey. Dec. 1953. Decl. Mar. 4, 1957. 84p. Contract AT-11-1-GEN-14. $0.50(OTS).

The effect of U burnup and Pu production on the regeneration factor, η, is studied for natural and slightly enriched U exposed to a maxwellian thermal neutron flux Analytic expressions are obtained for the U and Pu atomic densities, and the assumptions concerning these expressions are examined. Numerical data are obtained for the isotopic densities as a function of the time integrated flux $[= \int_0^t \varphi(t)dt]$ for several enrichments and three values of the total resonance absorption probability. Fission product poison data are presented and combined with the isotopic density data to obtain the variations of the regeneration factor, η. The effect of the variation in isotopic densities of fissionable material on the flux level when operating at constant power is investigated, and curves are presented which relate constant power operation to constant flux operation. Results indicate that for a given resonance absorption probability, curves of η versus flux-time corresponding to different enrichments all intersect in a common point. For neutrons with a maxwellian average temperature of 537°F the value of $\int_0^t \varphi(t)dt$ at this point is between 4 and 5×10^{21} cm^{-2}. The cause and significance of this effect are discussed. (auth)

10919 CF-56-9-97

Oak Ridge National Lab., Tenn.

CROSS SECTION PROGRAM AT ORNL. J. A. Harvey and J. L. Fowler. Sept. 18, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

10920 MLM-744

Mound Lab., Miamisburg, Ohio.

ABSORPTION CROSS SECTION OF ACTINIUM-227. Final Report. D. L. Timma and L. N. Russell. Aug. 21, 1952. Decl. Feb. 25, 1957. 8p. Contract AT-33-1-GEN-53. $0.25(OTS).

The neutron absorption cross section of Ac^{227}, as measured by the pile oscillation technique, was determined as 493 ± 35 barns. This value may be in error by about 7%. (C.H.)

NUCLEAR REACTORS

10921 ANL-4238

Argonne National Lab., Lemont, Ill.

REACTOR ENGINEERING DIVISION REPORT FOR THE PERIOD MARCH 1, 1948 THRU NOVEMBER 30, 1948. Feb. 3, 1949. Decl. Mar. 28, 1957. 67p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

Studies reported on equipment for circulating NaK in Experimental Breeder Reactor development include considerable information on wear of bearings in NaK. Blanket blocks of natural U provided with press-fitted finned cooling tubes have been fabricated and jacketed in deep-drawn Ni cans. Which have been welded shut and welded to the cooling tubes to form a gas tight envelope for the blocks. The steps in fabricating the cans and jacketing the blocks are described. A mock-up of a section of the blanket cup formed from the jacketed blocks has been thermal cycled to simulated operating conditions. The blanket cup safety device mock-up and safety rod mechanism are illustrated. One voltage drop method for detecting voids and meniocus in the NaK-filled fuel rods has been developed. Operating characteristics of electromagnetic pumps and a homopolar generator are reported. A NaK level indicator is described briefly. (G.Y.)

10922 ANL-5132(Del.)

Argonne National Lab., Lemont, Ill.

THE MANUFACTURE OF INTERNAL BLANKET AND FUEL BLANKET SLUGS FOR THE EXPERIMENTAL BREEDER REACTOR. R. E. Macherey and S. T. Zegler. Dec. 1953. Decl. with deletions Feb. 12, 1957. 61p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

Manufacturing methods and procedures for fabricating EBR internal and fuel blanket elements are presented in detail. (C.H.)

10923 ANL-5297

Argonne National Lab., Lemont, Ill.

REACTOR ENGINEERING DIVISION QUARTERLY RE-

PORT [FOR] APRIL 1, 1954 THROUGH JUNE 30, 1954. July 15, 1954. Decl. Mar. 28, 1957. 206p. Contract W-31-109-Eng-38. $33.30(ph OTS); $9.60(mf OTS).

Boiling Experimental Reactor. Revised core and fuel element designs have been established as a result of mockup tests on the removal of heat by natural air circulation at shutdown. A flow sheet for the BER is presented. Power Breeder Reactor. Major work on the EBR-II has been devoted to the design of a working model. Several plan views and perspective drawings are shown. Small Power Reactors. Further design factors for a small horizontal boiling power reactor are discussed. The reactor is primarily designed for operation in remote arctic regions at a power level of $5\frac{1}{2}$ Mw. Design Research and Development. A theoretical analysis was made of the effect of Pu^{239} produced in the BER on the reactor temperature coefficient. For the particular BER considered, it appears that the coefficient may become less negative as Pu^{239} builds up. It is also possible that the cross section temperature coefficient may be positive at reactor operating temperatures. Thermal and fast neutron fluxes were calculated for a given 20 Mw BER core, together with the γ flux. The values obtained for the thermal neutron and γ flux were used to determine the radioinduced heating in the Fe pressure vessel. The accuracy of modified diffusion theory (removal theory) for calculating the photon source distribution from neutron captures in the EBR shield was determined by comparison with measured values. Discrepancies noted are attributed to variations in the U^{238} fission rates and the use of point source geometry. New data have been obtained from a Monte Carlo calculation of neutron transmission and reflection by slabs of H_2O. The time dependent response of a reactor due to localized reactivity variations was treated for the case of low time rate of change where the response is measured some distance from the localized change. Amplitude responses and phase shifts were made as a function of driving function frequency for a series of short lifetimes. At very low frequencies, the amplitude response is not equally sensitive to all delay groups. The kinetics of a supercritical reactor with no sources is analyzed, and values of Σ, k_{ex}, and λ were calculated as a function of τ. Heat Transfer. A modified electrical–geometrical analogue has been designed to solve temperature distribution problems within fuel elements of complex geometry. Good accuracy for the device is reported. Factors affecting the precision of a given thermal analysis of EBR-II fuel elements are discussed, including uncertainty factors which are helpful in determining the worst possible effects of a number of thermal parameters. A heat transfer analysis was made for 5 different EBR-II fuel element designs. Three radiator-type and 2 pin-type elements were tested, and their comparative performance is summarized. Natural circulation tests were made to determine the steady–state relationship between maximum plate temperature and power density of shutdown heating. It is concluded that the reactor core will not be adequately cooled during shutdown when there is a loss of water from the core area. A pulse-heating test facility was designed to investigate the fundamentals of water expulsion from a Cu tube to which short pulses of a-c current are applied. Fuel Elements. The feasibility of fabricating the reference type EBR-II fuel element was investigated. This design consists of a hexagonal array of fuel tubes which consist of in-line fuel and blanket sections. An experimental ball tower apparatus was constructed for the fabrication of ball-type fuel elements. The results of 4 shotting runs are given. Data are collected on the water-corrosion of unclad U–Nb, U–Si, and U–Nb–Zr alloys at high temperatures. Other corrosion studies include corrosion-irradiation tests on Zircaloy-2-clad U fuel plates, the corrosion of Zircaloy-2 by superheated steam and moist and dry air, and the corrosion characteristics of plates with deliberate defects introduced in the clad. Structure Materials and Compound Tests. Four reactor pressure vessels, fabricated from Carilloy T-1 alloy steel, were tested to destruction by impact testing. Highly satisfactory results were achieved. Further tests are reported on the magnetic jack control rod drive unit described in the previous report. A preliminary investigation is under way to develop a void detector for boiling water systems. Most promising methods involve the absorption of γ or x rays, the dielectric constant of the steam-water mixture, the detection of boiling noise, and the transmission or scattering of visible light. Two types of liquid level indicators were examined for use in pressurized closed reactor systems. One instrument consists essentially of a resistance bridge circuit, whereas the second depends on the changing capacitance of an insulated probe immersed in the water tank. Coolants. The dynamic Na corrosion loop was operated at 400 and 500°C to determine the oxygen content of the Na as an oxide. For operating times of 8 to 55 hr, the content was found to vary between 0.001 and 0.0061 wt. % oxygen. Specific damage to diphenyl by fast neutron flux irradiation was studied. Design and Evaluation. Design details of the BER pressure vessel and control rod drive mechanisms for the BER and EBR-II are presented. (K.S.)

10924 BNL-60
Brookhaven National Lab., Upton, N. Y.
AN EXPERIMENTAL AND THEORETICAL STUDY OF THE SUBCRITICAL BNL REACTOR. J. Chernick, I. Kaplan, J. Kunstadter, V. Sailor, and C. Williams. June 15, 1950. Decl. Dec. 14, 1955. 109p. $0.55(OTS).

Experimental and theoretical studies of the subcritical BNL Reactor are summarized. Subcritical and exponential experiments are described. Experiments were also performed on transients and the effects of displacement of U. Theoretical consideration was given to lattice calculations, anisotropic effects, neutron leakage from the gap, reflector saving, and other reactor constants. (B.J.H.)

10925 BNL-2739
Brookhaven National Lab., Upton, N. Y.
LMFR PROGRESS LETTER FOR FEBRUARY 1956. F. T. Miles. Mar. 20, 1956. Decl. Feb. 18, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

10926 CF-52-8-148(Chap. 22)(Del.)
Oak Ridge School of Reactor Technology, Tenn.
REACTOR MATERIALS. CHAPTER 22. G. E. Evans. 1952. Decl. with deletions Feb. 14, 1957. 41p. $7.80 (ph OTS); $3.30(mf OTS).

A glossary of terms and abbreviations used in reactor technology is given. (M.H.R.)

10927 CF-55-5-156(Rev. 1)
Oak Ridge National Lab., Tenn.
HRT PROCESS FLOWSHEETS. Revision 1. F. C. Zapp. Aug. 25, 1955. Decl. Mar. 18, 1957. 20p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

10928 CF-55-8-175
Oak Ridge National Lab., Tenn.
ORR REACTOR PHYSICS: PROBLEM I, INFINITE SLAB MODEL. M. L. Nelson and N. F. Lansing. Aug. 31, 1955. Decl. Nov. 23, 1955. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

The components of the ORR are described in some detail with a view to obtaining, through calculation, a rough idea of neutron flux distribution. For the calculations, a very simple model was chosen, and some of the nuclear con-

stants were assumed the same as for a comparable fuel loading of the MTR. The flux distribution function was postulated to be separable in the 3 coordinate directions and flux distributions along each coordinate were obtained from a 2-group, 3-region analysis of infinite slabs each chosen so that the transverse configuration corresponds to that along one coordinate direction in the actual core. Properties and constants are tabulated and calculation procedures given. Numerical results were obtained from ORACLE. Fluxes were plotted and the beam current calculated. (T.R.H.)

10929 CF-56-2-35
Oak Ridge National Lab., Tenn.
HRT CORROSION SAMPLE HOLDERS. F. C. Zapp. Feb. 8, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Information concerning the location of HRT corrosion sample holders is tabulated. (B.J.H.)

10930 CF-56-2-99
Oak Ridge National Lab., Tenn.
HRP-CP: INSTRUMENT INTERLOCKS FOR THE HRT CHEMICAL PROCESSING PLANT. William L. Carter. Feb. 20, 1956. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Tables are given which describe the entire system of instrument interlocks for the HRT, indicate the affected instruments, and specify the conditions under which each instrument is to function. (B.J.H.)

10931 CF-56-2-159
Oak Ridge National Lab., Tenn.
FEEDWATER PUMP TESTS. Experiment No. V A-3. W. D. Burch and B. H. Hamling. Feb. 16, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The procedure proposed to test the HRT boiler feedwater pumps is given. A flow rate determination by measurement of level depletion in the deaerator-surge tank is included. (B.J.H.)

10932 CF-56-3-14
Oak Ridge National Lab., Tenn.
PRESSURE DROP CALIBRATION OF FUEL HEAT EXCHANGER. HRT Report No. I A(4)a. J. R. Engel. Mar. 6, 1956. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The pressure drop across the main fuel heat exchanger was measured as a function of flow rate using potable water. The pressure drop was also measured as a function of water temperature at a constant flow rate of 300 gpm. The ranges covered by the calibrations were 121 to 395 gpm for the first test and 7 to 250°C for the other. (auth)

10933 CF-56-3-138
Oak Ridge National Lab., Tenn.
HRP-CP: DESIGN OF CONDENSERS H-1 AND H-2 FOR HRT CHEMICAL PROCESSING PLANT. William L. Carter. Mar. 19, 1956. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

There are two condensers in the chemical plant low-pressure system which are used to dissipate heat from radioactive decay of fission products. One condenser is associated with each of the decay-storage vessels. Attenuation of β and γ radiation in the stored slurry of fuel solution and solids causes boiling and decomposition of D_2O. These vapors are passed through a recombiner for D_2 and O_2 recovery and then into the condenser from which the condensate is refluxed to the decay-storage vessel. The condenser is a conventional design of shell and tube heat exchanger. Some of the more pertinent data are summarized. (auth)

10934 CF-56-3-139
Oak Ridge National Lab., Tenn.
HRT-CP: DESIGN AND OPERATION OF THE RECOMBINERS FOR THE HRT CHEMICAL PROCESSING PLANT. William L. Carter. Mar. 8, 1956. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Radiolytic decomposition of heavy water in the decay-storage vessels of the HRT chemical processing plant requires that a recombiner be installed for D_2 and O_2 recovery. Decomposition gases will be diluted with steam, passed through an entrainment separator, superheated and passed over a platinized alumina catalyst. D_2 and O_2 will be present in stoichiometric amounts at concentrations well into the non-explosive region. After recombination the D_2O vapor will be totally condensed and refluxed to the decay-storage tank. The rate of water decomposition in the chemical plant is quite small when compared to the same reaction in the HRT core system. In fact sufficient steam will be produced by absorption of radiation in the fuel solution to dilute the combustibles below their explosive concentration. However, auxiliary heat has been provided for additional boil-up. To avoid poisoning the catalyst surface with condensed moisture, the recombiner is cradled in a mantle heated by 250 psig steam. Pertinent data on the recombiner design are tabulated. (auth)

10935 CF-56-4-27
Oak Ridge National Lab., Tenn.
HYDRAULIC TEST OF ORR DUMMY FUEL ELEMENT. E. S. Bettis. Apr. 5, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A hydraulic test was made on the 19-plate ORR fuel element in order to determine the pressure drop and flow distribution in the various channels. The test procedure is described in detail. Since the maximum difference in flow between any two channels in any one measurement amounted to 17%, it was concluded that the flow distribution in the element was satisfactory. (B.J.H.)

10936 CF-56-4-98
Oak Ridge National Lab., Tenn.
AIR-COOLED CONDENSER STEAM SYSTEM. HRT Engineering Test Procedure V 5. J. L. Gory and H. E. Williamson. Apr. 12, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Test procedures are given for determining the performance of the air-cooled condenser for the HRT. (B.J.H.)

10937 CF-56-4-140
Oak Ridge National Lab., Tenn.
PROCEDURE FOR PERFORMANCE TEST OF TURBINE CONDENSER AND ASSOCIATED EQUIPMENT. HRT Test V A 2, a, b, c. J. O. Kolb and R. E. Brooksbank. Apr. 18, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

10938 CF-56-4-194
Oak Ridge National Lab., Tenn.
HRT COOLING WATER SYSTEM EVALUATION — RESULTS OF HRT ENGINEERING TESTS VI A-2, 3, 4 5; B-1 a, b AND B-3. R. E. Brooksbank. Apr. 30, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The cooling water system has been tested and found to perform satisfactorily isothermally with the following important exceptions. The maximum observed flow to the fuel sample cooler was 3.8 gpm. The flowsheet specifies a maximum flow of 8 gpm. The maximum observed flow to the blanket sample cooler was 5.4 gpm. The required flow is 8 gpm. Larger inlet and exit lines to these coolers would permit more flow. It is recommended that these line sizes be increased from ½ in. to 1 in. Of less importance is that

the maximum flows to the fuel and blanket dump coolers were 4.1 and 21 gpm, respectively, compared to 10 and 30 gpm specified by the flowsheet. (auth)

10939 CF-56-5-58
Oak Ridge National Lab., Tenn.
HRP-CP: DESIGN CALCULATIONS FOR THE ENTRAINMENT SEPARATORS (S-2 AND S-3) FOR THE HRT CORE PROCESSING PLANT. William L. Carter. May 4, 1956. 25p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70 (mf OTS).

There are two entrainment separators in the HRT chemical processing plant to minimize particulate carry-over in the vapor from the decay-storage vessels. One separator is provided for each decay-storage vessel and is located in the vapor line near its exit from the vessel. De-entrained droplets collect at the bottom of the separator and drain back to the decay-storage vessel. To effect entrainment separation each unit employs three principles: centrifugation, impingement and the use of large contact surface in the form of wire mesh. An additional feature of the unit is a small condenser coil located at the vapor exit. The purpose of this coil is to provide reflux for flushing the inside of the separator thereby removing any crust or collection of solid matter which might impair the efficiency of the unit. This feature will greatly reduce or eliminate plugging tendencies. Pertinent characteristics, performance and operating information are given. (auth)

10940 CF-56-5-165
Oak Ridge National Lab., Tenn.
HRT MODIFIED PRESSURIZER DESIGN. Carlyle Michelson. May 25, 1956. 27p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The work done on modification of the HRT pressurizer design is outlined. HRT mock up operations as they pertain to the pressurizer are summarized, and establishment of the basic criteria for redesign is discussed. A general description of the internal configuration of the modified pressurizer and its mode of operation are presented. A flowsheet, conceptual drawing, and summary of design specifications are included. Calculations in the appendix justify values for the basic design parameters such as heater capacity, vaporization rate, water reserve, water level, and surge volumes. Appropriate references to unclassified literature are included. (auth)

10941 CF-56-5-167
Oak Ridge National Lab., Tenn.
EFFICIENCY OF FUEL SYSTEM ENTRAINMENT SEPARATOR DURING NORMAL OPERATION. HRT Report 11 A (7c)b. Work Period: September 27, 1955—November 25, 1955. Paul N. Haubenreich, J. O. Kolb, R. Van Winkel, and T. E. Haynes. May 25, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

During the preliminary testing of the fuel low-pressure system, the performance of the entrainment separator was checked by analyzing samples of the condensate from the condenser while boiling trisodium phosphate solution in the dump tanks. Samples were also taken from the dump tanks to establish the concentration there. Runs were made with tank levels between 95 and 600 liters, steam pressures from 30 to 100 psig and with and without cooling water passing through the coils in the entrainment separator. The decontamination factor, defined as the ratio of sodium concentration in the dump tank to that in the condensate, was in the range from 1000 to 10,000 in all runs. There was considerable scatter of the data and no clear relation between the decontamination factor and steam pressure or dump tank level was observed. The data are tabulated and the decontamination factor is plotted against steam pressure. (auth)

10942 CF-56-5-169
Oak Ridge National Lab., Tenn.
FEED PUMP AND PURGE WATER FLOW MEASUREMENTS. P. H. Harley. May 24, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Preliminary tests were made for a direct reading flow meter in the HRT feed pump. Results of these tests are shown graphically. (B.J.H.)

10943 CF-56-5-188
Oak Ridge National Lab., Tenn.
PRESSURE SHELL STRESS ANALYSIS HRR. M. C. Lawrence. May 25, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The calculated, total tangential stress for the HRR pressure vessel at steady state operation (500 Mw) is shown as a function of wall thickness and thermal shielding. The maximum stress, located at the inside shell surface, results from both internal pressure and irradiation heating. With thermal shields four inches thick or less, mounted inside the reactor shell, the lowest stress is more than 23,000 psi. For these conditions the shell irradiation heating varies from about 2100 to 20 Btu's/hr × in.^{3}, decreasing with penetration through the steel. Following five inches of shielding the irradiation heating of the pressure shell decreases very slowly and can be expected to average from 5 to 10 Btu's/hr × in.^{3} for the shell thickness considered. For these conditions stresses below 17,500 psi are calculated. For steel designation, A-212B, having an allowable working stress of 17,500 psi, a minimum wall thickness of about 3.3 inches is required. The curves are based upon the nuclear properties of iron and are applicable to low alloy-steel thermal shields and pressure vessels only. (auth)

10944 CF-56-6-142
Oak Ridge National Lab., Tenn.
GASEOUS RADIOACTIVITY FROM HRT FUEL STORAGE TANKS. R. F. Aven. June 22, 1956. 20p.. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A study has been made to determine the amount of gaseous radioactivity from the fuel solution when it is stored in the outer dump tanks. The main source of gaseous activity is the decay of the fission products Br^{83}, I^{133} and I^{135} into Kr^{83}, Xe^{133}, and Xe^{135} respectively. The radioactivity of the gases leaving the reflux condenser is much too great to be sent directly to the stack. For time periods after shutdown greater than 24 hours the gaseous activity may be found from graphs. The radiation release is directly proportional to long term power. Since the radiation release to the stack must be kept low (~20 curies/day or less) it is recommended that a recombiner be used in the fuel storage system to remove deuterium and oxygen from the gaseous stream leaving the outer dump tanks. The radioactive gases can then be either held indefinitely in the low pressure system or sent through the main charcoal beds, before release to the stack. The only safe alternative to the recombiner is to hold the fuel solution in the inner dump tanks for a period of about 10 days for 5 Mw power (11 days for 10 Mw). The rate of radiation release after this period is less than 20 curies/day and can be sent directly to the stack. (auth)

10945 CF-56-8-187
Oak Ridge National Lab., Tenn.
CALCULATIONS APPLICABLE TO HRT-BP DEVELOPMENT: LOOP P-1. P. G. Lafyatis. Aug. 28, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Preliminary calculations were made for the operation of P-1 loop of the HRT. Heat exchanger characteristics and P-1 volumes are tabulated. The initial and final volumes of gas and liquid in the loop are also given. (B.J.H.)

10946 CF-56-8-199
Oak Ridge National Lab., Tenn.
ORR PROJECT PROGRESS REPORT NO. 19. J. P. Gill and T. E. Cole. Aug. 15, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Progress on the construction of the ORR is briefly reviewed. (For preceding period see ORNL-CF-56-5-190.) (B.J.H.)

10947 CF-56-9-111
Oak Ridge National Lab., Tenn.
DESIGN SECTION MONTHLY REPORT [FOR] SEPTEMBER 1956. W. R Gall and M. I. Lundin. Sept. 27, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Progress in the development of HRT components is reported. (D.E.B.)

10948 CF-56-10-77
Oak Ridge National Lab., Tenn.
HEAT-POWER CYCLES AND PRIME MOVERS FOR NUCLEAR POWER PLANTS. R. C. Robertson. Oct. 10, 1956. 88p. Contract [W-7405-Eng-26]. $13.80(ph OTS); $4.80(mf OTS).

A general discussion of the subject is given. Main topics included are basic principles, performance calculations for steam and gas turbine cycles, comparison of the steam and gas cycles, saturated steam cycle performance and design considerations, supercritical pressure steam cycle, heat power cycles for atomic plants, performance trends for steam plants, power cycle costs and trends, and optimum cycles for nuclear plants. (M.H.R.)

10949 CF-56-10-104
Oak Ridge National Lab., Tenn.
PRELIMINARY REPORT ON INSTRUMENTATION FOR ORR HRP LOOP HN-1. R. A. Lorenz. Oct. 23, 1956. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

Loop instrumentation functions are summarized. Component failures and temperature control are discussed. Loop instrumentation and an automatic data logger and alarm scanner are suggested. Radiation monitoring recommendations for ORR loop HN-1 are included in an appendix. (M.H.R.)

10950 CF-56-11-57
Oak Ridge National Lab., Tenn.
EFFECT OF HRT-CP COOLING WATER REQUIREMENTS ON THE REACTOR CIRCULATING LOOP. R. E. Brooksbank. Nov. 15, 1956. 5p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Because of the problem of possible chloride-induced stress corrosion cracking of the 347 S.S. cooling water jackets in the HRT, the reactor cooling water circuit will be modified to provide a closed demineralized water loop. Primary cooling water will be obtained from the cooling tower. This report presents the results obtained from a study of the HRT-CP cooling water system and the effect on the future reactor circulating loop circuit. It is proposed that the HRT-CP have a demineralized cooling water circuit independent of the reactor system. This system should be similar to the reactor scheme which is to be installed during December 1956 and would contain a heat exchanger, surge tank and pump. The HRT-CP at the present time receives a 15 gpm cooling water supply from the reactor cooling tower circulating pump. The proposed chemical plant dissolver scheme will employ an additional 16 to 27 gpm requirement. The maximum flow that can be expected during normal operation at any one time is 32 gpm which includes the current measured flows required by the chemical plant high-pressure system and the quantity required by the new system. The projected HRT-CP cooling water use if the blanket processing plant is considered is 64 gpm. A breakdown of the HRT-CP cooling water requirements is presented. (auth)

10951 CF-56-11-58
Oak Ridge National Lab., Tenn.
HRT SOURCE HANDLING FACILITIES. C. A. Burchsted and C. L. Segaser. Nov. 15, 1956. 5p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A portable shield is described which may be used to transport the HRT antimony-beryllium source from the LITR to the HRT reactor site, and to store the source during critical experiments preliminary to bringing the reactor to power. The shield consists of a 16-in. diam. lead pig (the existing HRT fluid sample carrier) centered in a 50-in. diam. × 42-in. deep steel tank filled with borated water. (auth)

10952 CF-56-11-105
Oak Ridge National Lab., Tenn.
ORR PROJECT PROGRESS REPORT NO. 20. J. P. Gill and T. E. Cole. Nov. 15, 1956. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Progress on the construction of the ORNL Research Reactor is briefly described. Discussions are also included on component fabrication, experimental facilities, and control. (For preceding period see ORNL-CF-56-8-199.) (B.J.H.)

10953 HW-24137
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
REACTIVITY EFFECTS OF VERY INHOMOGENEOUS ENRICHED LOADINGS. H. Neumann. Feb. 24, 1953. Decl. Feb. 19, 1957. 21p. Contract W-31-109-Eng-52. $0.25 (OTS).

The reactivity effects of one and of two highly enriched fuel rods placed into a natural uranium lattice thermal pile are investigated according to the small source model. This model seems well suited to problems involving very inhomogeneous loadings. Assuming moderator constants and blackness values for the pile fuel elements, the resonance escape probabilities for those neutrons arising from fissions in the enriched rods are calculated to be, in the case of a single enriched rod loaded centrally into the pile, 0.855, and for the case of two enriched rods loaded centrally into the pile, 0.868. (auth)

10954 HW-46483
Hanford Atomic Products Operation, Richland, Wash.
PRPR CONTROL REQUIREMENTS. PART I. J. R. Triplett. Nov. 14, 1956. Decl. Feb. 8, 1957. 8p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The basic requirements of the PRP Reactor control system are briefly discussed. (B.J.H.)

10955 KAPL-M-EDL-113
Knolls Atomic Power Lab., Schenectady, N. Y.
REPLACEMENT ACTUATOR SURGE POT BAFFLE TEST S2G. D. J. Oakley and J. J. Stewart. Apr. 17, 1956. 23p. Contract [W-31-109-eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Various designs of baffles were tested in order to determine the most effective combination for preventing sodium splashing into the annulus around the control drive shaft at the top of the actuator surge pot. The recommended combination consists of a diverger-type baffle installed on the shaft and a deflector ring placed 2.5 inches above. (auth)

10956 KAPL-M-EJ-4
Knolls Atomic Power Lab., Schenectady, N. Y.
STATUS REPORT ON S3G/S4G UNIT CELL MECHANICAL

ANALYSIS. E. B. Johansson. Aug. 24, 1956. 19p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The unit cell design has been checked for compliance with the SAR reactor design basis. The major results are given and the method of analysis is outlined. Calculations are compared with experimental results. The major problems considered are mechanical shock and fluid pressure acting on the ZR cans and channels. A summary of future work is included. (auth)

10957 KAPL-M-FDJ-4

Knolls Atomic Power Lab., Schenectady, N. Y.

SAR PRIMARY COOLANT IMPURITY ACTIVATION AND DEMINERALIZER SHIELDING. Frank D. Judge. [1956]. 23p. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

The activity of various primary coolant impurities are required to determine the dose rate contributions from primary coolant pipes and equipment after shutdown. In addition, the demineralizer concentrates the activated ion impurities and it is shown that a $1\frac{1}{2}$ in. lead shield is required to reduce the dose rate on the demineralizer surface to 200 mr/hr 15 minutes after shutdown (lower reactor compartment dose rate specification). Equilibrium impurity concentrations estimated from STR Mark I tests, a 6 GPM demineralizer flow rate, and a 15,000 GPM total primary coolant flow rate were assumed. (auth)

10958 KAPL-M-HH-7

Knolls Atomic Power Lab., Schenectady, N. Y.

VARIATIONAL AND SUCCESSIVE COLLISION RESULTS FOR THERMAL UTILIZATION IN SLAB LATTICES. Henry Hurwitz, Jr. Sept. 18, 1956. 10p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

The relationship between the variational method and the successive collision method for calculating thermal utilization in slab lattices with nonscattering fuel is discussed. Also expressions are derived for the probability that a neutron produced uniformly in the moderator will suffer its next collision in the moderator. This quantity is all that is required for evaluating the variational expression with constant trial function. (auth)

10959 KAPL-M-JAM-1

Knolls Atomic Power Lab., Schenectady, N. Y.

THERMOCOUPLE IN-PILE CALIBRATION FOR S3G. J. A. McCann. Jan. 4, 1956. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A small piece of a pure metal is assembled into an S3G temperature-instrumented fuel element so that it surrounds the bead of the thermocouple. At the melting point of the pure metal, a nearly isothermal condition occurs, decreasing the measurement error resulting from the large temperature gradients in the element. The stresses of differential thermal expansion between the thermocouple and the fuel element are also relieved at melting. A single-point temperature calibration of the thermocouple is thus achieved at the freezing point of the pure metal, during operation of the reactor. Results of a simple experiment indicate the sensitivity, reproducibility, and also feasibility of this proposal. A future work program is outlined. (auth)

10960 KAPL-M-JLV-1

Knolls Atomic Power Lab., Schenectady, N. Y.

EXAMINATION OF LUMMUS CONDENSER TUBING. C. F. Barrett, Jr., J. L. Van Ullen, and Lyall Zickrick. Feb. 9, 1956. 13p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Four tubes of Admiralty metal in the Lummus off-hull condenser leaked because of a combination of exposure to mercury and residual stress. 70/30 cupronickel alloy is relatively immune in the same environment. (auth)

10961 KAPL-M-JPS-3

Knolls Atomic Power Lab., Schenectady, N. Y.

NEUTRON SOURCES FOR REACTORS. Jan P. Simpelaar. Mar. 12, 1956. 11p. Contract W-31-109-Eng-52. $1.80 (ph OTS); $1.80(mf OTS).

The optimum source material for a long endurance, mobile power reactor is shown to be Ra-Be. Two other alpha emitters, Po and Pu, are rejected—Po for its short half-life, and Pu for its inability to withstand a high thermal neutron flux. Photoneutron sources are also rejected, primarily because they are not dependable and require auxiliary sources. (auth)

10962 KAPL-M-MJA-2

Knolls Atomic Power Lab., Schenectady, N. Y.

NEUTRON LEAKAGE THROUGH AIR AND ALUMINUM FILLED DUCTS. Marvin J. Abrams. Mar. 10, 1954. Decl. Mar. 14, 1957. 44p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

The literature was surveyed to select a method of calculating neutron leakage through empty ducts which penetrate a reactor shield. The simplest and best verified approach was found to be the Simons–Clifford Theory, an inverse square theory in which empirical coefficients describe the reflection properties of the duct elbows. The applicability of this theory is checked and corrections are made. The resulting leaking flux equation can be used to calculate the primary shield blister size and to check neutron leakage through the secondary shield penetrations (e.g., through steam risers, downcomers, etc.). (M.H.R.)

10963 KAPL-M-RCD-34

Knolls Atomic Power Lab., Schenectady, N. Y.

MODEL 17A FUEL ROD. Robert C. Hammers. Feb. 8, 1956. 21p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

All welds made on the twelve assemblies were of the strength required by Engineering. However, due to the deviations from the recommendations for good joint design as set forth by MIL-W-6858 and the Americal Welding Society, all joints failed to meet the minimum strength requirements called for in MIL-W-6858, with the exception of the upper retainer to perforated tube weld. (auth)

10964 KAPL-M-S3G-RE-502

Knolls Atomic Power Lab., Schenectady, N. Y.

MECHANICAL TESTS OF THE SAR $7\frac{1}{8}$ INCH STAINLESS STEEL UNIT CELL. Test No. 502. A. B. Briggs. Sept. 5, 1956. 11p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Mechanical tests were performed on the SAR $7\frac{1}{8}$ inch stainless steel unit cell; included were tension, compression and bending tests, plus both internal and external pressurizing. (auth)

10965 KAPL-M-S3G-RE-507

Knolls Atomic Power Lab., Schenectady, N. Y.

ENDURANCE TESTS ON SAR ZIRCALOY 3 FUEL ELEMENT WELDS; TEST NUMBER 507. A. B. Briggs. Sept. 5, 1956. 12p. SAR Project. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

An endurance-testing machine was constructed incorporating a high temperature, high pressure autoclave for tests on SAR fuel element welds. Five specimens were tested to failure and results compared. (auth)

10966 KT-188

Massachusetts Inst. of Tech., Oak Ridge, Tenn. Engineering Practice School.

DESIGN OF BOILING THORIUM BLANKET. C. J. Billerbeck, W. S. Delicate, J. Farquhar, III, F. D. Miraldi, P. L Perry, A. D. Rossin, and A. W. Wendling. Dec. 10, 1954. Decl. Mar. 4, 1957. 63p. For Carbide and Carbon Chemicals Co. [K-25 Plant. Contract W-7405-eng-26, Subcontract 70.] $10.80(ph OTS); $3.90(mf OTS).

10967 NAA-SR-81
North American Aviation, Inc., Downey, Calif.
TWO-GROUP EQUATIONS FOR MULTI-REGION REACTORS. E. R. Cohen and M. A. Greenfield. Aug. 28, 1950. Decl. Feb. 26, 1957. 21p. Contract AT-11-1-GEN-8. $0.25(OTS).

The age-theory of the diffusion of neutrons in a multiplying medium was replaced by an equivalent two-group description that may be more readily applied to multi-region calculations. The two-group equations, which exhibit essential differences from the straight-forward two-group theory involving mono-energetic diffusion of neutrons, were then used to develop a perturbation theory which yields the inhour equation in a somewhat generalized form. (auth)

10968 NAA-SR-112
North American Aviation, Inc., Downey, Calif.
REACTOR SELF-SHIELDING. M. Neustadt and E. R. Cohen. Apr. 20, 1951. Decl. Mar. 2, 1957. 54p. Contract AT-11-1-GEN-8. $9.30(ph OTS); $3.60(mf OTS).

The leakage flux of neutrons and γ rays from the reflector of a spherical pile is calculated. This leakage flux is of interest because it determines the attenuation which must be furnished by the external shield. The calculation, given here, is accomplished by following the neutrons from collision to collision. It is compared to a "discrete age theory" calculation, to an ordinary "age theory" calculation and to a Christy-Wheeler kernel calculation. Calculations are carried out for a spherical reactor having a non-neutron absorbing moderator and reflector of the same substance. The neutron leakage from the reflector, above energy E, per neutron produced is found. The leakage based on the "straight-ahead" calculation and on the "discrete age theory" calculation differ by no more than a factor of two over the range $0.5 < E < 2.0$ Mev. The γ-ray leakage from the reflector per gamma ray produced in the core is calculated for a spherical reactor having a moderator and reflector of the same material. The γ-ray leakage is given as a function of $\mu\gamma$ R over a wide range of $\mu\gamma$ t where $\mu\gamma$ is the γ ray absorption coefficient in the material, R is radius of the core and t is the thickness of the reflector. (auth)

10969 NAA-SR-113
North American Aviation, Inc., Downey, Calif.
SHIELDING FOR THE LOW POWER RESEARCH REACTOR. M. Neustadt. Feb. 20, 1951. Decl. Feb. 27, 1957. 12p. Contract AT-11-1-GEN-8. $0.25(OTS).

The radiation shield required for the LPRR was determined to be 34 inches of MO concrete. Neutrons, primary and secondary gamma rays, and self-shielding effects were considered. (D.E.B.)

10970 NAA-SR-138(Pt. II)
North American Aviation, Inc., Downey, Calif.
INTRA-CELL NEUTRON DENSITIES IN NATURAL URANIUM–D_2O LATTICES. PART II. D. C. Woods and A. T. Biehl. June 1, 1953. Decl. Feb. 27, 1957. 48p. Contract AT-11-1-GEN-8. $0.35(OTS).

Results are presented of measurements of intra-cell neutron density distributions in natural uranium–D_2O lattices as the rod diameters are varied from $^3/_4$ to 2 inches and the cell spacing from 3 to 12 inches. By comparing the saturated activities of gold and europium foils, it was also possible to obtain a measure of the effective temperature of the thermal neutrons in the fuel rods of the various lattices relative to that in the moderator. Interpretation of the results indicates the following: the thermal neutron density distribution inside the rods as well as in the D_2O moderator, as calculated from diffusion theory, is not a good approximation to the distribution measured experimentally; and the neutron temperature increase due to selective diffusion of the neutrons into the uranium rods exceeds that due to the Wigner-Wilkins effect. (auth)

10971 CF-53-2-78
Oak Ridge National Lab., Tenn.
CRITICAL CONCENTRATION IN THE ISHR AS A FUNCTION OF TEMPERATURE. M. Tobias and P. N. Haubenreich. Feb. 9, 1953. Decl. Apr. 2, 1957. 8p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The critical concentration in the ISHR was determined for temperatures between 100°C and 275°C at 25°C intervals. Calculations were made assuming 3% voids in the core, and for no voids in the core. Shell thicknesses of $^1/_8$ inch and $^1/_4$ inch were considered. The blanket was assumed to contain 1000 g Th per liter. The reactor was taken to be "clean" with no multiplication in the blanket. The two-group method described in CF-51-10-110 was employed at 100°C and 250°C. For the other temperatures, the critical concentration was estimated by adjusting the concentration for a bare reactor by assuming that the percentage deviation between the bare reactor and reflected reactor critical concentrations was a linear function of temperature. The percentage deviation varied between 15% and 34% for the cases considered. It is believed that the concentrations calculated in this way are within a few percent of concentrations which would be obtained by performing a complete two-group analysis at each temperature. (auth)

10972 NAA-SR-209
North American Aviation, Inc., Downey, Calif.
BUCKLING MEASUREMENTS OF THERMAL NEUTRONS IN NATURAL URANIUM – D_2O SQUARE LATTICES. S. W. Kash. Dec. 1, 1952. Decl. Feb. 27, 1957. 16p. Contract AT-11-1-GEN-8. $0.25(OTS).

The buckling of the thermal neutron distribution for 22 natural U–D_2O square lattices has been experimentally determined. A brief description of the apparatus used is given. The computational methods employed and the corrections applied to the data are presented in detail. (auth)

10973 ORNL-1711
Oak Ridge National Lab., Tenn.
PRESENT STATUS OF THE INVESTIGATION OF AQUEOUS SOLUTIONS SUITABLE FOR USE IN A THORIUM BREEDER BLANKET. M. H. Lietzke and W. L. Marshall. June 7, 1954. Decl. Feb. 28, 1957. 12p. Contract [W-7405-Eng-26]. $0.25(OTS).

Several aqueous Th systems were investigated in relation to their applicability as breeder blankets. Difficulty with the hydrolysis of $Th(NO_3)_4$ in $Th(NO_3)_4-H_2O$ systems was encountered near 230°C. The stability range was improved by the addition of excess HNO_3. In all cases, the precipitates formed did not redissolve upon cooling to room temperature. Unfavorable solubility and viscosity characteristics were associated with $ThO_2-H_3PO_4-H_2O$ systems. The $ThO_2-H_3PO_4-HNO_3$ system is stable at elevated temperatures only in the presence of concentrated H_3PO_4. This system and the $Th(NO_3)_4-HNO_3$ system are considered most suitable for further research. (K.S.)

10974 ORNL-1746
Oak Ridge National Lab., Tenn.
DETERMINATION OF CORROSION PRODUCTS AND ADDITIVES IN HOMOGENEOUS REACTOR FUEL. A. D. Horton, P. F. Thomason, and M. T. Kelley. Sept. 24, 1954. Decl. Mar. 2, 1957. 26p. Contract W-7405-eng-26. $0.30(OTS).

Ni and Mn have been separated from U in 6M HCl solutions of HRT fuel by ion exchange on a Dowex A-1 chloride-form resin column and determined simultaneously by polarography in NH_3-NH_4Cl electrolyte. The relative

standard deviation of this method is 3.7% for Ni at a concentration of 10 μg per ml and 6.5% for Mn at a concentration of 2.0 μg per ml. Ni and Mn have been separated from U in 0.005 M H_2SO_4 solutions of HRT fuel by ion exchange on a Dowex A-1 sulfate-form resin column and determined simultaneously by polarography in NH_3-$(NH_4)_2SO_4$ electrolyte. The relative standard deviation of this method is 5% for Ni at a concentration of 2.0 μg per ml and 3.4% for Mn at a concentration of 2.0 μg per ml. The lower limit of sensitivity of the methods is 0.5 μg of Ni or Mn per ml of the solution used for the polarographic analysis. Cu in HRT fuel has been determined by polarography of a 0.25M EDTA solution of the fuel at pH 8.5. The half-wave potential of Cu in this medium is approximately −0.55 volt vs. the S.C.E. By this method, Cu in HRT fuel has been determined in concentrations of 1 to 10 μg per ml. The relative standard deviation of the method is 0.65% for Cu at a concentration of 10 μg per ml. Zr has been separated from U in 6M HCl solutions of HRT fuel by ion exchange on a Dowex A-1 chloride-form resin column and determined spectrophotometrically as the Thoron complex at a wavelength of 555 mμ. The range of the method is 10 to 100 μg of Zr, and the relative standard deviation is 4.2% for this range. (auth)

10975 TID-5304

Martin (Glenn L.) Co., Baltimore.

FEASIBILITY OF INCREASING MTR FLUX LEVEL. June 30, 1955. Decl. Mar. 1, 1957. 39p. Contract AT(11-1)-355. (ER-7474). $7.80(ph OTS); $3.30(mf OTS).

An increase in MTR power by a factor of 1.5 appears feasible through the substitution of a tube-type fuel assembly for the presently used plate-type assembly. No alterations to the present heat removal systems and shielding would be required. The suggested fuel assembly would contain 110 tubes arranged in a 10 × 11 array and having the external dimensions of the presently used assembly. Heat transfer, fluid flow, and reactivity studies associated with the suggested assembly modification are given. (D.E.B.)

10976 TID-10038

North American Aviation, Inc., Downey, Calif.

PARAMETER SURVEY FOR DEPLETED AND NATURAL URANIUM LATTICES. E. Richard Cohen and R. H. Sehnert. Apr. 7, 1952. Decl. Mar. 26, 1957. 70p. Contract AT-11-1-GEN-8. $0.45(OTS).

The properties of three distinct lattice systems were studied, using uranium (0.1, 0.3, and 0.711% U^{235}) as fuel and absorber; graphite, heavy or light water as moderator; and light water as coolant. Data are presented for given thicknesses of the water cooling annulus, showing the thermal utilization, resonance escape probability, diffusion length, Fermi age and multiplication constant as a function of the fuel rod radius and cell size. (auth)

10977 TID-10052

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

SUMMARY REPORT ON THE REACTOR HAZARDS ASSOCIATED WITH THE DANGER COEFFICIENT TEST FACILITY (DCTF). Apr. 30, 1951. Decl. Mar. 20, 1957. 187p. Contract AT-11-1-GEN-14. $0.80(OTS).

The DCTF is a homogeneous critical assembly using $UO_2(NO_3)_2$ designed to determine the reactivity characteristics of fuel assemblies. The design, planned operating procedures, and safety considerations for the assembly are given. Extensive weather and population studies of the proposed facility site are included. (D.E.B.)

10978 TID-10082

Oak Ridge National Lab., Tenn.

HRE DESIGN MANUAL. C. L. Segaser and F. C. Zapp. Nov. 18, 1952. Decl. Apr. 5, 1957. 413p. Contract W-7405-eng-26. $2.40(OTS).

All pertinent mechanical and process design data considered relevant to the operation and maintenance of the HRE are compiled. (auth)

10979 WAPD-2

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

LECTURE SERIES ON ELEMENTARY REACTOR THEORY SEPTEMBER–DECEMBER, 1949. Sidney Siegel. Decl. Feb. 6, 1957. 139p. Contract AT-11-1-GEN-14. $0.65 (OTS).

Discussions are included on the critical condition in fissile assemblies, various neutron reactions, the life history of a neutron in a reactor, neutron diffusion, characteristics and solutions to the pile equation, neutron slowing down, reactor kinetics, and control. (B.J.H.)

10980 WAPD-87

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

ON THE RATIO OF MAXIMUM TO AVERAGE POWER DENSITY. Milton J. Galper. July 1953. Decl. Mar. 4, 1957. 33p. Contract AT-11-1-gen-14. $0.30(OTS).

A method is described for calculating the ratio of maximum to average radial power density for a heterogeneous cylindrical core having two concentric regions with different material bucklings. It is shown that the results can be made independent of core size. Several techniques for varying the buckling in the two regions are discussed, and as an example the results obtained for one particular design are included. (auth)

10981 KAPL-M-S3G-RE-423

Knolls Atomic Power Lab., Schenectady, N. Y.

INLET PLENUM TEST; TEST NUMBER 423. C. E. Ockert. July 18, 1956. 48p. SAR Project. Contract [W-31-109-Eng-52]. $9.30(ph OTS); $3.60(mf OTS).

A flow study was made on a transparent, half-scale model of the inlet plenum region of the S3G Power Unit. Visual observations were made at various flow rates using injections of dye slugs, or air bubbles, to trace the flow patterns. Slow motion movies and stop watch observations were used to study the time distribution of entry of a dye slug to the various first pass standpipes in connection with cold water reactivity accident studies. In addition, streak photographs were taken with air bubble injection to record the flow patterns. Pilot tubes were employed in an unsuccessful effort to obtain reliable data for the quantitative distribution of the coolant to the 12 first pass standpipes. The results of the test indicated a complicated four eddy pattern in the plenum region. It was concluded from this test and also in consideration of the results of the air flow analogue, that the proposed inlet baffle would be undesirable, since the improvement of flow distribution was not worth the pressure drop penalty (5 psi). (auth)

10982 WAPD-EM-2

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE EFFECT OF FINS OR SPACERS ON HEAT TRANSFER, LOCAL BOILING AND PLATE STRESSES IN A SYSTEM WITH WATER FLOWING TURBULENTLY BETWEEN PARALLEL PLATES. J. T. Stiefel. Jan. 6, 1956. Decl. Mar. 25, 1957. 29p. $4.80(ph OTS); $2.70(mf OTS).

10983 WAPD-FE-770

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].

PROPOSAL FOR THE FIRST UO_2 HT-1 TEST. J. H. Bach. May 2, 1955. Decl. Mar. 5, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

A test is proposed to check the effects of fuel rod warp-

ing accompanied by partial blocking of the coolant channel to the extent that one fuel rod will contact two neighboring rods. (F.S.)

10984 WAPD-M-92
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
REVISED SPECIFICATIONS FOR MTR-HT-1 DEFECT ASSEMBLIES. Feb. 15, 1955. Decl. Mar. 18, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Descriptions are given of fuel element test assemblies consisting of bundles of elements with the center element intentionally defected so that failure will occur soon after the test unit is brought up to reactor operating conditions, in a manner approximating the conditions of failure that might be expected in the ultimate failure of a PWR fuel element in service. The tests are meant to demonstrate that the failure of one fuel element in an assembly will not necessarily cause the failure of neighboring elements. (B.J.H.)

10985 NAA-SR-179(Rev.)
North American Aviation, Inc., Downey, Calif.
LOW ENRICHMENT URANIUM THERMAL REACTORS FOR THE PRODUCTION OF PLUTONIUM AND USEFUL POWER. E. F. Weisner. July 7, 1952. Decl. Mar. 7, 1957. 103p. Contract AT-11-1-GEN-8. $0.65(OTS).

A study has been made of reactor systems for the production of Pu and useful power, using fuel enriched in the U^{235} isotope (between 1 and 2% ratio of U^{235} to U^{238}). Also, the use of highly (93.4%) enriched fuel in conjunction with natural U (spiked systems) has been investigated. Graphite and H_2O were considered most suitable as moderators; stainless steel and Zr, as cladding materials for the fuel, and various liquid metals as coolants, with the principal emphasis on Na. It is concluded that a pressurized water system and a Na cooled, graphite moderated system are entirely feasible at this time and have many inherent advantages over systems which are designed to use natural U fuel only. The use of slightly enriched fuel allows a design with a relatively high conversion ratio and one which can also use materials and temperatures which are favorable to useful power generation. The outlet temperature of the pressurized water system is between 450° and 500°F; while, from the sodium cooled system, it can be between 750° and 900°F, depending on the scheme of cooling employed. (auth)

10986 WAPD-MRP-44(Rev.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD MARCH 1, 1954 TO APRIL 22, 1954. Decl. Mar. 14, 1957. 66p. Contract AT-11-1-gen-14. $0.50(OTS).

A set of reference core thermal and hydraulic design parameters has been selected. The preliminary core configuration and fuel rod design are presented. Pure water pumping rates for a given reactivity change were studied for a homogeneous chemical poison type of control as a function of U enrichment, Xe override, and the degree of control required at the end of life. Some melting and fabrication techniques for U–Mo and U–Zr alloys and fuel rods are described. Corrosion results are reported for LiOH on steel at 600°F, the effects of neutron irradiation on hydrazine hydrate, the time for discontinuous failure of U–Mo, U–Mo–Nb, and U–Nb alloys in 650°F H_2O and the effect of H_2 on the corrosion of U–Mo alloys. A program was initiated for the study of H_2O corrosion several Zr alloys. Investigations of the phase transformations in U–Mo and U–Nb alloys have begun, together with the mechanical properties of Zircaloy-2. Flux calculations were made for a typical lattice configuration, using spherical harmonics. (K.S.)

10987 WAPD-MRP-47(Rev.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD JULY 15, 1954 TO AUGUST 26, 1954. Decl. Mar. 13, 1957. 93p. Contract AT-11-1-gen-14. $0.60(OTS).

Examination has been completed of a corrosion-tested brazed joint between pieces of 347 stainless steel plate. Au–Cu alloy is the most promising material tested. Preliminary reactor designs have been developed for several seed core configurations. Attempts are being made to devise a core which will increase the power generated in the blanket over that developed in the core elements. Further testing of fabrication techniques for Zircaloy-clad U–Mo rods is in progress. Butt welding of the fuel element and closures appears most feasible at this time. Other tests were conducted on the fabrication of UO_2 and U–Si fuel rods, with moderate success. The thermal and hydraulic properties of PWR core designs are being investigated for several fuel element assemblies. Tentative burnout data at 2000 psia have been obtained for a typical tubular specimen. Further investigations of the phase transformations in U–Nb and U–Mo alloys were carried out. Water corrosion testing is reported on Zircaloy-clad U–Mo, U–Nb, and U–Si fuel elements. Tensile tests of U–Mo alloys were conducted. The performance of several corrosion inhibitors for Cr and Mo steel has been evaluated, and other tests are continuing. Fission product removal methods were studied by measuring the decontamination of stainless coupons by the addition of decontaminants to loop water. Periodic acid yielded high test decontamination factors. Five fission products have been analyzed for the total water content of the test loop, the collected crud, the ion-exchange resin, and the steel coupons. The thermal stability of boric acid solutions as a chemical poison was confirmed, and no difference in the heat transfer coefficient was noted between water and 0.15M boric acid solutions. The corrosive effects of boric acid solution on several structural materials are reported. A group of U–Mo fuel element samples was exposed in the process water of the MTR. The results of corrosion tests on these samples are summarized. Similar data are given for unclad U–Si alloys. More than 20 core design criticality studies have been completed with a one-dimensional spatial simulator. A computer study of the time required for Xe burnout has been completed. (K.S.)

10988 NAA-SR-274
North American Aviation, Inc., Downey, Calif.
SODIUM GRAPHITE REACTOR. Quarterly Progress Report [for] March–May 1953. G. M. Inman, ed. Sept. 25, 1953. Decl. Mar. 18, 1957. 116p. Contract AT-11-1-gen-8. $0.65(OTS).

Studies were continued relating to the nuclear physics and engineering characteristics of the sodium cooled, graphite moderated reactor for power production. Considerable effort was devoted to study of the fuel element from the standpoint of neutron performance, metallurgical properties, fabrication problems and factors affecting the heat transfer. A general parameter study was performed to establish the reactor characteristics leading to minimum cost power. These characteristics included such quantities as lattice spacing, fuel element uranium cross section, number of rods per fuel element cluster, and initial excess reactivity. Separate investigations were carried out in connection with problems of shield design,

sodium leakage detection and fabrication of double walled coolant tubes. These studies were conducted with particular reference to a design comprising a graphite stack threaded by vertical coolant tubes manifolded at top and bottom. Alternate designs making use of different moderators and different means of manifolding the sodium flow were investigated. Relative advantages were noted and comparisons made. As a separate design study, the sodium graphite reactor was investigated for possible application as a package power plant. The characteristics of such a unit capable of 14 megawatt electrical output were established in some detail. (auth)

10989 WAPD-PA-228

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PWR PRIMARY COOLANT VOLUME SURGES. Interim Report Based on the First Two Groups of Simulator Studies. T. J. Walker. July 1955. Decl. Mar. 19, 1957. 40p. Contract AT-11-1-GEN-14. $6.30(ph OTS); $3.00(mf OTS).

The methods used and results obtained from PWR simulator runs with respect to volume and temperature transients of the primary coolant system are summarized. All major primary plant components were simulated with the exception of the pressurizer. Volume surges at constant pressure are given as computed for the most recent simulator runs. (auth)

10990 NAA-SR-1568

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

REACTOR SAFETY QUARTERLY PROGRESS REPORT [FOR] AUGUST–OCTOBER 1955. Stephen H. Fitch and Norman E. Huston, eds. May 15, 1956. Decl. Mar. 27, 1957. 30p. Contract AT-11-1-GEN-8. $0.35(OTS).

Calculations of the amount of corrosion of safety device components by boron trifluoride have been made. They are based on the chemical and pressure measurements of the NAA-13 boron trifluoride irradiation test capsules. The indications are that corrosion is not extensive enough to influence either the operating characteristics or the integrity of the reactor safety device. An unsuccessful attempt was made to conduct the MTR-NAA-9 solder plug irradiation experiment. Wiring difficulties caused premature melting of the safety fuse and no data were obtained from the experiment. Further thermal transient data have been obtained on thin slabs which approximate the cross sections of experimental safety device trigger elements. A vacuum chamber was utilized to eliminate convection cooling. At reduced pressures, the thermal relaxation time is approximately twice as long as that obtained at atmospheric pressure. Work is continuing, to obtain lower pressures and to investigate the effect of radiation which may be causing additional uncertainties. Electrical analogs of three thin slabs have been fabricated and tested. Relaxation time data have been obtained for these analogs and for the NAA-8 fuse assembly analog using improved circuitry. Relaxation times are: analog of NAA-8 fuse, $\theta = 1.27$ second, and analogs of all the thin slabs, equivalent $\theta \cong 1.05$ second. The values agree quite well with analytical work; but agreement has not been achieved with thermal measurements. It is expected that discrepancies may be resolved by improvements on the NAA-8 fuse analog and refined techniques on the thin slab thermal measurements. The Mark III zirconium receiver pressure test has been completed. The data show that the 20-thousandths of an inch wall is insufficient to retain the boron trifluoride at elevated temperatures. Further experiments will be made with heavier wall zirconium. (For preceding period see NAA-SR-1560.) (B.J.H.)

10991 WAPD-RDa-60

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PWR REFERENCE BLANKET FUEL ROD. N. J. Palladino. Aug. 3, 1955. Decl. Mar. 8, 1957. 3p. PWR Project. Contract [AT-11-1-GEN-14]. $1.80(ph OTS); $1.80(mf OTS).

10992 WAPD-ReC(A)-35

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PRESSURE DROP FOR SHORT FLAT PLATE FUEL ELEMENTS. J. E. Zerbe and H. S. Jacket. Nov. 23, 1953. Decl. Mar. 8, 1957. 11p. Contract [AT-11-1-GEN-14]. $3.30(ph OTS); $2.40(mf OTS).

Pressure drop tests were made for a fuel element design consisting of banks of short flat parallel plates. The banks were arranged in-line or staggered in the direction of flow. The plates were one in. long in the direction of flow, and $\frac{1}{8}$ in. thick, and had a lateral spacing of .09 in. The results were in the form of combined joint loss coefficients that covered both the expansion loss and contraction loss at each junction of the plates. For in-line arrangements the joint loss coefficient, K_j, was reduced from approximately .50 with axial plate clearances of .19 and .29 inches to approximately .25 with a .09 inch axial spacing. For staggered arrangements the coefficient remained essentially constant with spacing. The value was .53. The calculated joint loss coefficient for full expansion and contraction based on area ratio was .64. (auth)

10993 WAPD-TH-40

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PROPOSAL FOR IN-PILE BURNOUT TEST AT MTR. H. Dobel and S. J. Green. Mar. 11, 1955. Decl. Mar. 9, 1957. 41p. Contract AT-11-1-gen-14. $6.30(ph OTS); $3.00(mf OTS).

A proposal is presented for an in-pile nucleate boiling heat transfer burnout test at the MTR. The object of the test was to confirm burnout results previously obtained by the use of electrically-heated test sections of pressure tubes. The proposed test conditions and test section design are summarized. (C.H.)

10994 WAPD-TM-1

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

SUBPOWER RANGE START-UP TRANSIENTS. E. G. Good. Mar. 1956. Decl. Mar. 8, 1957. 117p. Contract AT-11-1-GEN-14. $0.70(OTS).

A parametric study of the reactor start-up transient in the subpower range has been conducted. The general-purpose analog computer was used to simulate the reactor kinetics equations. The nuclear parameters have been varied over a sufficiently wide range to encompass the values normally encountered in any thermal reactor. Besides showing the shape of the subpower range transients, the curves obtained provide the necessary initial conditions for extending the transients into the power range for any particular reactor. The wide range of linear reactivity insertion rates used should cover the range of control rod withdrawal rates encountered in both normal start-up and start-up accidents. A method is outlined for determining the conditions that will result if the reactivity insertion is stopped at any time in the transient. Hence, the curves may be used to study the start-up transients both when the control rods are withdrawn continuously and when the control rods are stopped at any time in the transient. The significance of each of the parameters varied is discussed. Also, the simulator techniques peculiar to this particular study are outlined. (auth)

PARTICLE ACCELERATORS

10995 CRD-T4A-64
California Research and Development Co., Livermore, Calif.
EXPECTED HEAT LOADS IN A THORIUM FUELED LATTICE. C. C. Old. Nov. 8, 1951. Decl. Feb. 15, 1957. 7p. (LWS-24191). $1.80(ph OTS); $1.80(mf OTS).

10996 LRL-118
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
MTA PROJECT TARGET AND PROCESS SECTION QUARTERLY PROGRESS REPORT [FOR] JULY THROUGH SEPTEMBER 1953. Apr. 1954. Decl. Mar. 2, 1957. 68p. Contract AT(11-1)-74. $10.80(ph OTS); $3.90(mf OTS).

MTA target studies included data calculations on a H_2O-cooled, U block target; neutron yield comparisons between cooled (NaK) and uncooled Be–U and U–U target systems under deuteron bombardment; and isotopic purity investigation in both primary and secondary U targets. The proton-induced reactions of Th and the spallation of both U^{238} and Th^{232} were studied. Nuclear physics work included experimentation on inelastic cross section measurement and measurement of the intensity profile of the deuterons from stripped He^3 ions. Experimental diffusion studies were done concerning the thermal and In resonance energy neutron flux from a source at one end of a small, H_2O-filled cavity and in two MTA target lattice models. The flux perturbation caused by a wire neutron detector was also observed. Theoretical calculations were made on the neutron yield and heat release from U targets under deuteron bombardment. Diffusion study calculations included slow neutron behavior around a hollow duct in water and preliminary production characteristics of light-metal primary–U secondary target–lattice concepts. Target design development work centered mostly on evaluation of the over-all economics of a wide range of target concepts and design parameters. Five additional preliminary target designs were conceived. Base case target product costs have been estimated for a large range of currents, energies, and production rates. A study has been made of power recovery feasibility in an MTA target. (auth)

10997 UCRL-1124
California. Univ., Berkeley. Radiation Lab.
III. TWO-GROUP APPROXIMATION TO ABSORBING LATTICES. Harold Brown. Feb. 12, 1951. Decl. Mar. 21, 1957. 4p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

The mathematics of the approximation is developed for the case of a H_2O-cooled, graphite-moderated matrix. (D.E.B.)

10998 UCRL-1135
California. Univ., Berkeley. Radiation Lab.
V. MULTIPLYING LATTICES OF THE HOLLOW TYPE. Harold Brown. Feb. 14, 1951. Decl. Mar. 21, 1957. 4p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

A two-group approximation of the fission ratios and Pu production in an infinite hollow matrix is developed. (D.E.B.)

10999 UCRL-1817
California. Univ., Berkeley. Radiation Lab.
MTA MARK II ACCELERATOR, DRIFT TUBE AND RF BEAM GEOMETRY PERTURBING EFFECT OF BEAM IN A-12. Myron Good. Apr. 8, 1952. Decl. Mar. 5, 1957. 5p. Contract [W-7405-Eng-48]. $1.80(ph OTS); $1.80(mf OTS).

RADIATION ABSORPTION AND SCATTERING

11000 CF-56-8-219
Oak Ridge National Lab., Tenn.
THE GAMMA ENERGY-ABSORPTION COEFFICIENT. Lloyd G. Alexander. Aug. 8, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Equations are developed for the γ energy absorption coefficient. Numerical results are tabulated and shown graphically. (B.J.H.)

11001 KAPL-M-JPS-4
Knolls Atomic Power Lab., Schenectady, N. Y.
GAMMA RAY ENERGY ABSORPTION KERNELS. Jan. P. Simpelaar. Mar. 23, 1956. 35p. Contract W-31-109-Eng-52. $6.30(ph OTS); $3.00(mf OTS).

A gamma ray multigroup method was used to determine energy absorption kernels of the form $\sigma e^{-\sigma x}$. Kernels are found for both core and reflector regions. Several gamma ray spectrums representing full power operation and fission product decay gammas at various times after shutdown are used. (auth)

RADIATION EFFECTS

11002 TID-10066
North American Aviation, Inc., Downey, Calif.
LONG TERM IRRADIATION OF NUCLEAR FUELS. E. R. Cohen and R. H. Sehnert. Sept. 15, 1954. Decl. Mar. 1, 1957. 49p. Contract AT-11-1-gen-8. $0.40(OTS).

With the objective of reducing fuel fabrication and chemical processing and thus the cost of power from nuclear reactors, it has been proposed that nuclear fuels be exposed to irradiations as high as 15,000 mwd/ton. Such exposures correspond to almost complete burnout of the original U^{235} contents of uranium fuels. A study is made of the changes in heavy isotope concentrations as well as the build-up of fission product poisons in fuel elements and the corresponding changes in local and effective reactivity as a function of irradiation exposure. Original uranium enrichments of 0.7115, 0.978, 1.46, and 1.94 weight per cent of U^{235} have been studied. The fuel is considered loaded into reactors operating at 300°K with initial conversion ratios varying in increments of 0.1 from 0.6 to 1.0. Changes in reactivity, fractional burnup of U^{235} and buildup of Pu^{239}, Pu^{240}, and Pu^{241} are presented over the range of parameters and interpreted for specific reactor objectives. (auth)

SHIELDING

11003 BMI-1039
Battelle Memorial Inst., Columbus, Ohio.
HIGH-BORON STEELS FOR REACTOR SHIELDING. Henry A. Saller, John T. Stacy, and Herbert L. Klebanow. Sept. 27, 1955. Decl. Feb. 12, 1957. 28p. Contract W-7405-eng-92. $0.30(OTS).

Several carbon, alloy, and stainless steels containing from 1.0 to 2.5 wt. % boron, generally about 1.25 wt. % boron, were surveyed to determine their ability in the wrought condition to meet minimum design requirements of a 0.2% offset yield strength of 100,000 to 150,000 psi and a tensile elongation of 8 to 10%. An age-hardenable 17-7 stainless steel containing 1.25 wt. % boron most nearly satisfied these specifications. With additions of 1.25 wt. % boron, the following steels possessed insufficient ductility: SAE 1010, 1020, 1030, and 4140, Type 431 stainless, Armco 17-4PH, and Armco 17-7PH. Precipitation-hardening steels A-286 and V-2-B, with 1.25 wt. % boron, were sufficiently ductile but could not be aged to the required yield strength by the commercially recommended heat treatments used. (auth)

11004 KAPL-M-LPP-1
Knolls Atomic Power Lab., Schenectady, N. Y.
THE CAUSE OF EXCESS PRESSURE IN B_4C SHIELDING CANS. Leonard P. Pepkowitz. Aug. 7, 1952. Decl. Mar.

14, 1957. 8p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The pressure increase observed during the heat cycling tests is caused by sorbed water in the vibrating mix. At elevated temperatures the steam reacts with the free carbon to form methane or with the stainless steel container to form hydrogen. The water can be removed by heating above 200°C. The best conditions found were 400°C with pumping. Above 500°C the B_2O_3 fuses which cements the vibrating mix together. Therefore heating above 500°C should be done, if feasible, only after the mix is in the shield cans. Care must be taken to prevent resorbtion of moisture after the drying treatment. As much as 0.1 wt. % of water will be resorbed in 2 hours at 20% relative humidity. (auth)

URANIUM AND URANIUM COMPOUNDS

11005 BMI-800(Del.)
Battelle Memorial Inst., Columbus, Ohio.
THE ROLLING OF URANIUM. H. A. Saller, J. R. Keeler, and R. J. Donley. Dec. 30, 1952. Decl. with deletions Feb. 12, 1957. 38p. Contract W-7405-eng-92. $6.30(ph OTS); $3.00(mf OTS).

Uranium rods and strips were rolled under various rolling conditions and the roll-separating force measured. The roll load increased with decreasing rolling temperature so that at 300°C, the loads were three to five times as great as at 600°C. The rolling speed between 61 and 122 fpm had little effect on the roll load, and the effect of the initial structure of the metal was also small. A temperature rise of 6 to 12°C per pass occurred in the work on rolling at 600°C and a 35 to 44°C rise occurred at 300°C. Rods rolled at 400°C and lower had cracks and other rolling defects. Mild steel rolled at 1030°C (1900°F) produced nearly the same roll load as uranium at 600°C, and the spread of uranium and steel strip also was nearly the same. The specific resistance to deformation or yield strength of uranium during rolling was calculated by the method of Orowan and varied from 26,000 psi at 600°C to 81,000 psi at 300°C. (auth)

11006 K-406
Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART IV. Dixon Callihan, D. F. Cronin, J. K. Fox, R. L. Macklin, and J. W. Morfitt. Nov. 28, 1949. Decl. Mar. 15, 1957. 34p. Contract [W-7405-eng-26]. $0.30(OTS).

Some exploratory experiments are reported describing the conditions under which enriched uranium, contained in aqueous solutions of uranyl fluoride, becomes critical in two right cylindrical reactors having parallel axes. Values of the height of solution and the critical mass have been obtained for reactors with diameters ranging from 5 to 20 inches. Data were obtained at four chemical concentrations corresponding to $H:U^{235}$ atomic ratios varying between 30 and 330 and with reactor separations up to 50 cm. In some of the experiments, the reactors were submerged in a water bath while others were done with no reflector. The critical mass of an unenclosed two reactor system was found to depend upon the distance between the components when they were separated even as much as 50 cm, however, the mass in each was then more than 90% of that required to make it singly critical. Interposition of water between two reactors reduced their interaction due to the attenuation of the neutron flux by hydrogen. Two water enclosed reactors which could be made singly critical when approximately equilateral were found to be effectively isolated when separated 15 cm or more. Two which could not be made individually critical were isolated by a separation of a few centimeters. Those which were singly critical at heights large compared to their diameters showed apparent interaction at more than 20 cm spacing. The effect is attributed to the equivalence of a few interaction neutrons in the two component system and the relatively large quantity of uranium which must be placed at the end of long reactors to produce small increases in reactivity. The smallest mass accumulated at criticality in these experiments was 680 gm U^{235} contained in each of two water enclosed reactors ten inches in diameter with sides in contact, each filled to a height of 16.9 cm. The chemical concentration of the fuel corresponded to an $H:U^{235}$ atomic ratio of 329. (auth)

11007 K-643
Carbide and Carbon Chemicals Div. K-25 [Plant], Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART V. Dixon Callihan, D. F. Cronin, J. K Fox, and J. W. Morfitt. June 30, 1950. Decl. Mar. 19, 1957. 57p. Contract W-7405-eng-26. $0.40(OTS).

An investigation has been made of the conditions under which aqueous solutions of enriched uranyl nitrate become critical in right cylindrical reactors. A comparison was made of the neutron reflectivity of stainless steel, bismuth subcarbonate both dry and as a water slurry, aqueous solutions of natural uranyl nitrate and phosphoric acid. The materials tested, with the exception of dry bismuth subcarbonate, were about as effective reflectors as water. Dry bismuth subcarbonate was considerably less effective. The free nitrogen content of the nitrate solution and the addition of phosphoric acid and metallic bismuth to the reactor core were among the variables studied. Nitric and phosphoric acid when introduced into the core material were effectively mild poisons when compared with aqueous solutions at the same hydrogen to U^{235} ratio. Bismuth was introduced into the core as an array of aluminum clad bismuth rods. The critical mass of this array was only slightly less than that measured when the bismuth was replaced by a similar array of voids. (auth)

11008 KS-219
Carbide and Carbon Chemicals Co. [K-25 Plant], Oak Ridge, Tenn.
MINIMUM CRITICAL MASS OF U-235 FOR SEVERAL VALUES OF THE U-235 ASSAY. R. Gwin and H. F. Henry. Aug. 17, 1951. Decl. Mar. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

11009 NYO-3789
Naval Research Lab., Washington, D. C.
A COMPARISON OF THE ELASTIC CONSTANTS FOR ROLLED URANIUM BARS AT 25°C AND 300°C. E. W. Kammer, I. Vigness, and L. C. Cardinal. Oct. 15, 1953. Decl. Feb. 27, 1957. 16p. Contract AT(30-1)-1344. (NRL-4264). $0.25(OTS).

A measurement of the Young's modulus, shear modulus, and Poisson's ratio is described using ultrasonic pulses of approximately 3 mc/sec. The specimens used were in the form of "uncanned" bars about 8 inches long and 1 inch in diameter, some of which had detailed fabrication histories available. An electric furnace with helium atmosphere is described which permitted ultrasonic tests to be made at 300°C. In general, the elastic constants were reduced by about 15% at 300°C, compared to room temperature values. (auth)

11010 NYO-3790
Naval Research Lab., Washington, D. C.
THE EFFECT OF HEAT TREATMENT ON INTERNAL

FRICTION IN ROLLED URANIUM BARS. S. D. Hart. Oct. 15, 1953. Decl. Feb. 27, 1957. 12p. Contract AT(30-1)-1344. (NRL-4265). $0.25(OTS).

Rolled uranium bars of various heat treatments during fabrication were vibrated in their first flexural mode (about 1500 cps) and measurements made of the logarithmic decrement when the excitation was removed. This logarithmic decrement is taken as a measure of the internal friction. There is some indication of decrease in decrement with increase in the rolling process temperature, although heat treatments above the alpha-beta transition do not seem to produce any systematic influence upon the decrement. The decrement was greatly increased by mechanical shock or moderate static stress applied to the specimens. Recovery from this treatment was nearly complete after approximately 1000 hours. It is concluded that internal friction is too sensitive to handling and conditions of test to be used as an industrial control for the preparation of uniform rolled uranium bars. (auth)

11011 SEP-216

Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y.

THE ALPHA-BETA TRANSFORMATION TEMPERATURE AND GRAIN GROWTH CHARACTERISTICS OF HOT-PRESSED URANIUM. R. Resnick, L. Frank, and R. W. Balluffi. Oct. 28, 1953. Decl. Mar. 1, 1957. 15p. (YE53-517). $3.30(ph OTS); $2.40(mf OTS).

The alpha to beta transformation temperature of U, prepared either by hot pressing U powder of a 175 μ particle size or UH_4 powder of a 15 μ particle size, was found to correspond to that of the wrought metal within expected experimental error. The transformation temperature of cold compacted 175 μ particle size U powder was also found to agree with that of the wrought material. Measurements of grain growth in hot-pressed U and UH_4 possessing an initial grain size of 8 to 12 μ as a result of the hot pressing operation indicate that little or no normal grain growth takes place during annealing in the alpha temperature range. (auth)

11012 WAPD-PMM-120

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

RÉSUMÉ OF URANIUM OXIDE DATA-I. Z. M. Shapiro. May 2, 1955. Decl. Mar. 15, 1957. 33p. $6.30(ph OTS); $3.00(mf OTS).

The work on UO_2 was directed toward the utilization of the bulk material as a fuel for the PWR core. It was anticipated that UO_2 elements would be zircaloy-2 clad rods 0.4 to 0.5 in. in diam by 10 to 12 in. long filled with oxide having a density of 90% theoretical or better. The cladding would be 0.027 to 0.03 in. thick. Approximately 80,000 of these rods arranged in suitable clusters would be contained in the "blanket" of the core. The determination of pertinent properties and evaluation of the material as a bulk fuel, and the development of suitable methods of fabrication including such work as is necessary to determine the quality of the method were considered. (J.E.D.)